Duke Power®
A Duke Energy Company

M. S. Tuckman
Executive Vice President
Nuclear Generation

Duke Power
526 South Church St. EC07H
Charlotte, NC 28202
P. O. Box 1006 EC07H
Charlotte, NC 28201-1006
(704) 382-2200 OFFICE
(704) 382-4360 FAX

October 28, 2002

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Subject: Comments on the U.S.N.R.C Safety Evaluation with Open Items Related to the License Renewal of McGuire Nuclear Station, Units 1 & 2 and Catawba Nuclear Station, Units 1 & 2

Docket Nos. 50-369, 50-370, 50-413 and 50-414

By letter dated June 13, 2001, Duke Energy Corporation (Duke) submitted an Application to Renew the Facility Operating Licenses of McGuire Nuclear Station and Catawba Nuclear Station (Application). The Application contains the technical information required by 10 CFR Part 54 and the Supplement to the Final Safety Analysis Report (FSAR) for each station as required by §54.21(d). In a letter dated August 14, 2002, the NRC staff provided Duke a copy of the "Safety Evaluation Report with Open Items Related to the License Renewal of McGuire Nuclear Station, Units 1 and 2, Catawba Nuclear Station, Units 1 and 2." This staff letter requested that Duke review the enclosed safety evaluation report (SER), verify its accuracy, provide comments, and respond to the open and confirmatory items by October 27, 2002. (By letter dated August 29, 2002, the staff stated that this date is not a business day and revised the response due date to October 28, 2002.)

In a letter dated October 2, 2002, Duke provided an interim response that was intended to expedite the staff's completion of its review and to support the staff in its presentation to the Advisory Committee on Reactor Safeguards that occurred on October 8, 2002. Duke committed to provide responses to remaining open and confirmatory items identified in the SER, as well as Duke comments on the SER and revisions to the UFSAR Supplements for McGuire and Catawba.

Attachment 1 contains the Duke responses to the Open Items for Reactor Coolant System related items. These items were discussed informally with the staff on September 17, 2002. Some of the informal responses have been revised based on these discussions with the staff.

A085

Attachment 2 contains the Duke responses to the Open Items for Mechanical System related items. These items were discussed informally with the staff on September 18, 2002. Some of the informal responses have been revised based on these discussions with the staff.

Attachment 3 contains information on three topics. First, information is provided to supplement our previous responses to Open Item 2.5-1 previously provided by Duke letter dated October 2, 2002. Second information is provided to supplement our previous responses to Open Items 3.5-1 and 3.5-3. Finally, information is provided in response to a staff electronic communication dated October 23, 2002 concerning steam generator divider plates and pressurizer surge and spray nozzle thermal sleeves.

During the license renewal review process, many commitments were made by Duke to revise the UFSAR Supplements for McGuire and Catawba that were contained in Appendix A of the Application. Attachments 4 and 5 contain revised UFSAR Supplements for McGuire and Catawba, respectively, that incorporate all of the commitments that had been made to revise these two UFSAR Supplements. Following issuance of the renewed operating licenses for each unit of McGuire and Catawba, Duke will incorporate each station's supplement into its respective UFSAR on or before the next required update.

Duke has provided to the license renewal project manager, by separate communication, comments on the SER with open items (including comments on the revised excerpt provided by NRC letter dated October 19, 2002). Should there be any questions by the staff on these comments, Duke is prepared to discuss them further.

Finally, in a letter dated October 19, 2002, the staff provided a revised excerpt from SER with open items and request for additional information to complete the staff's review of the McGuire and Catawba license renewal application. Duke will provide its responses to the two staff requests, one on Inaccessible Non-EQ Medium-Voltage Cable Aging Management Program and one on aging effects for condenser circulating water system expansion joints, by November 6, 2002.

If there are any questions, please contact Bob Gill at (704) 382-3339.

Very truly yours,

M. S. Tuckman

M. S. Tuckman

Attachments:

Affidavit

M. S. Tuckman, being duly sworn, states that he is Executive Vice President, Nuclear Generation Department, Duke Energy Corporation; that he is authorized on the part of said Corporation to sign and file with the U. S. Nuclear Regulatory Commission the attached response to the Safety Evaluation with Open Items Related to the License Renewal of McGuire Nuclear Station, Units 1 & 2 and Catawba Nuclear Station, Units 1 & 2 , Docket Nos. 50-369, 50-370, 50-413 and 50-414, and that all the statements and matters set forth herein are true and correct to the best of his knowledge and belief. To the extent that these statements are not based on his personal knowledge, they are based on information provided by Duke employees and/or consultants. Such information has been reviewed in accordance with Duke Energy Corporation practice and is believed to be reliable.

M. S. Tuckman

M. S. Tuckman, Executive Vice President
Duke Energy Corporation

Subscribed and sworn to before me this 28TH day of OCTOBER 2002.

Mary P. Nelms

Notary Public

My Commission Expires:

JAN 22, 2006

xc: (w/ Attachment)

L. A. Reyes
Regional Administrator, Region II
U. S. Nuclear Regulatory Commission
Atlanta Federal Center
61 Forsyth Street, SW, Suite 23T85
Atlanta, GA 30303

D. B. Matthews
Director, Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Senior NRC Resident Inspector
McGuire Nuclear Station

Senior NRC Resident Inspector
Catawba Nuclear Station

C. P. Patel
Senior Project Manager
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

P. T. Kuo
Program Director, License Renewal and Environmental Impacts
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

R. L. Franovich
Senior Project Manager
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

R. E. Martin
Senior Project Manager
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

xc: (w/ Attachment)

Henry J. Porter
Assistant Director, Division of Waste Management
Bureau of Land & Waste Management
S.C. Department of Health and Environmental Control
2600 Bull St.
Columbia, SC 29201

R. M. Fry
Director, Division of Radiation Protection
North Carolina Department of Environment, Health, and Natural Resources
3825 Barrett Drive
Raleigh, NC 27609

North Carolina Municipal Power Agency Number 1
1427 Meadowwood Boulevard
P.O. Box 29513
Raleigh, NC 27626

North Carolina Electric Membership Corporation
P.O. Box 27306
Raleigh, NC 27611

Piedmont Municipal Power Agency
121 Village Drive
Greer, SC 29651

Saluda River Electric Cooperative, Inc.
P. O. Box 929
Laurens, SC 29360

Attachment 1

Response to
McGuire Units 1 & 2 and Catawba Units 1 & 2
Safety Evaluation Report with Open Items

Reactor Coolant System Related Items

New Open Item 3.0.3.10.2-1 The staff believes that volumetric examination of a sample of small-bore Class-1 piping is needed to demonstrate that the effects of aging are being adequately managed. Volumetric examination techniques provide a demonstrated capability and a proven industry record to permit detection and sizing of significant cracking and flaws in piping weld and base material. The sample of affected welds selected for inspection should be based upon piping geometry, pipe size and flow conditions, and the inspection should be performed by qualified personnel using approved station procedures.

Duke Response to New Open Item 3.0.3.10.2-1
New Open Item 3.0.3.10.2-1 was discussed with the staff during a meeting held on September 17, 2002. During the this meeting, the staff provided the following additional expectations:

> Applicant is to identify whether small bore piping (< 4 inches) with butt weld (socket welds excluded) could be susceptible to stress corrosion cracking or thermal fatigue cracking resulting from turbulent penetration or thermal stratification. Provide bases for conclusion. Volumetric examination of critical susceptible locations.

As discussed in Appendix B page B.3.20-5 of the Application, Duke has proposed that aging of small bore piping (piping less than 4-inch NPS) be managed by Risk-Informed Inservice Inspection (RI-ISI) requirements. The risk-informed approach is based on WCAP 14572 Revision 1-NP-A and consists of the following two essential elements: (1) a degradation mechanism evaluation is performed to assess the failure potential of the piping under consideration, and (2) a consequence evaluation is performed to assess the impact on plant risk in the event of a piping failure.

Duke submitted a request for relief, pursuant to 10 CFR 50.55a (g), to obtain staff approval of RI-ISI for McGuire Units 1 and 2 on June 26, 2001 (just after the submittal of the license renewal application on June 13, 2001). Supplemental information in support of this request relief was provided by Duke letters dated January 11, 2002 and March 15, 2002.

RI-ISI will allow Duke to perform volumetric examinations of certain risk significant small bore piping. Inspection locations are based on damage mechanism and consequences. Damage mechanisms considered in RI-ISI include: fatigue, stress corrosion cracking, and flow assisted corrosion/wastage. The fatigue model assumes that all failures by this mechanism result from preexisting flaws. Inputs to the model are sufficiently flexible to address low cycle fatigue attributable to normal plant transients, high cycle thermal fatigue (resulting, for example, from stratification of fluids and turbulent penetration), and high cycle vibrational fatigue. Duke letter dated January 11, 2002 to the staff identifies the specific degradation mechanisms considered for the Reactor Coolant System (NC) (entries on pages 3 of 37 and 4 of 37 of the attachment).

The NRC staff approved the use of RI-ISI on McGuire Units 1 and 2 by safety evaluation provided by letter dated June 12, 2002.

Risk informed assessment has not been completed for Catawba. Catawba is expected to have similar results and therefore should have a sample of small bore piping that will be volumetrically examined due to future implementation of risk-informed methods.

For the reasons stated above, Duke believes that the staff concern is effectively addressed by the recently approved RI-ISI program for McGuire. A similar RI-ISI program will be implemented at Catawba which will also address volumetric examinations of a sample of small-bore Class 1 piping.

New Open Item 3.0.3.10.2-2 In October 2000, a through-wall crack was identified in the reactor vessel hot leg piping at V. C. Summer. Specifically, the crack was located in the first weld between the reactor vessel nozzle and the "A" loop hot leg piping, approximately 3 feet from the reactor vessel and 7 degrees clockwise from the top dead center of the weld (as viewed from the centerline of the reactor vessel). The weld was fabricated from Alloy 82/182 material. The failure mode was determined to be primary water stress corrosion cracking and the root cause of the cracking was attributed to the presence of high residual stresses resulting from extensive repairs of the subject weld. The staff requests the applicant to identify the locations in the McGuire and Catawba RCS piping that contain welds fabricated from Alloy 82/182 material. Additionally, the staff requests the applicant to describe the actions it plans to take to address this operating experience as it applies to McGuire and Catawba.

Duke Response to New Open Item 3.0.3.10.2-2
New Open Item 3.0.3.10.2-2 contains two specific staff requests. In response to the first request, the following is a list the locations in the McGuire and Catawba reactor coolant system piping that contain welds fabricated from Alloy 82/182 material:

- Pressurizer surge, spray, relief, and safety nozzles weld buildup (Table 3.1-1, page 3.1-9, row 2 of the Application)
- Reactor vessel, primary inlet and outlet nozzles, buttering and welds (Table 3.1-1, page 3.1-11, row 3 of the Application)
- Steam Generator primary nozzle welds (Table 3.1-1, page 3.1-22, row 3, of the Application)
- Auxiliary feedwater nozzle safe end (Alloy 600 Safe End) (Table 3.1-1, page 3.1-25, row 4)

In response to the second request, the following actions have been taken to address the V.C. Summer operating experience as it applies to McGuire and Catawba. As part of EPRI MRP Alloy 600 ITG, the Alloy 82/182 Weld Integrity Inspection Committee was formed. Duke participated in this committee, which recommended that demonstrations be performed to document the capability of automated ultrasonic examination techniques for detecting inside surface-connected flaws in smooth bore nozzle configurations.

The VC Summer hot leg nozzle weld was a field weld (not a machined smooth bore nozzle configuration as is the design at Catawba and McGuire). If the weld surface is not smooth good contact cannot be maintained between the UT probe and the weld, which causes inaccurate results.

The geometry of the VC Summer weld was identified as a contributing factor in UT not identifying some of the part depth axial flaws in the hot leg nozzle weld.

Vendors that perform these examinations (Framatome ANP for Duke) performed examinations on a mock-up to demonstrate the effectiveness of their examination techniques. Framatome ANP results were found to be acceptable. The results are documented in EPRI 1006225, "Automated Ultrasonic Inside Surface Examinations of Reactor Coolant System Alloy 82/182 Nozzle Welds Performed in Spring 2001." McGuire Unit 1 results of 10 year ISI nozzle to safe weld examinations are documented in EPRI 1006225 (page 4-3). McGuire Unit 2 and the Catawba Units 1 and 2will have similar inspections during their 10 year ISI.

New Open Item 3.1.2.2.2-1 Under the Monitoring and Trending element of the Pressurizer Spray Head Examination, the applicant stated that a visual examination (VT-3) would be performed, and that no actions are taken as part of this program to trend inspection or test results. However, the staff's position is that VT-3 examinations may not be capable of detecting cracks that may occur in the pressurizer spray head. The staff therefore requests that the applicant amend the Pressurizer Spray Head Examination to state that VT-1 examination methods, which are capable of detecting and resolving cracks in the pressurizer spray heads, will be used for the one-time inspection. The scope of this open items includes the potential need to revise the acceptance criteria for this program and the FSAR Supplement summary description.

Duke Response to New Open Item 3.1.2.2.2-1
In response to New Open Item 3.1.2.2.2-1, Duke agrees to revise the visual inspection of the pressurizer spray head to VT-1. Acceptance criteria will be in accordance with ASME Section XI.

The UFSAR Supplements will be revised to reflect the VT-1 visual inspection and acceptance criteria.

New Open Item 3.1.3.2.2-1 The staff reviewed the surveillance capsule schedules in Tables B.3.26-1 and B.3.26-2 of the LRA. For McGuire 1, capsule "W" is a stand-by capsule and would be withdrawn at a fluence that is significantly above the equivalent of 60 years. The applicant needs to remove this capsule and place it in storage to prevent further exposure and preserve its ability to provide meaningful metallurgical data. For Catawba 2, capsule "U" is a stand-by capsule. It appears to the staff that this capsule should be inserted in the reactor vessel and begin to accumulate fluences in an operating environment for data collection purposes. The staff believes that the applicant should place all pulled capsules in storage so that they may be saved for future use. In addition, after the applicant has pulled all the capsules, it should use alternative dosimetry to monitor neutron fluence during the period of extended operation. The applicant needs to discuss its plans for this capsule with the staff.

Duke Response to New Open Item 3.1.3.2.2-1
The staff has raised concerns associated with McGuire Unit 1, Capsule W and Catawba Unit 2, Capsule U. For McGuire Unit 1, Capsule W is a standby capsule and is being used to support a sister plant. Capsule W has the same weld material as the limiting material of the sister plant. Capsule W contains material which is not the limiting material for McGuire Unit 1. Capsule W is not necessary to adhere to 10CFR50 Appendix H or E-185 withdrawal schedule for McGuire Unit 1. Presently, it is planned to withdraw Capsule W during EOC 18, which will cause it to have a little less than 2 times the EOL surface fluence of McGuire Unit 1.

The treatment of Capsule W will be clarified by revising the McGuire UFSAR Supplement Table 18.0-2 to read as follows:

McGuire Reactor Vessel Capsule Withdrawal Schedule

Unit	Capsule	Withdrawal End of Cycle (EOC)	Projected EOC Date	Estimated Fluence ($n/cm^2 \times 10^{19}$)	Reference
Unit 1	U	1	2/24/84	0.405	WCAP-10786
Unit 1	X	5	10/12/88	1.50[a]	WCAP-12354
Unit 1	V	8	3/12/93	2.08 [b][c]	WCAP-13949
Unit 1	Y	11	2/14/97	2.86 [d]	WCAP-14993
Unit 1 (dosimetry analysis & storage) Note 1	Z	8	3/12/93	2.38	WCAP-13949
Unit 1	W	18	4/5/04	4.52	Note 2
Ex-vessel Cavity Dosimetry	N/A	12	5/29/98	1.58	WCAP-15253
Unit 2	V	1	1/25/85	0.323	WCAP-11029
Unit 2	X	5	7/5/89	1.47[a]	WCAP-12556
Unit 2	U	7	1/9/92	2.04 [b][c]	WCAP-13516
Unit 2	W	10	4/5/96	3.07 [d]	WCAP-14799
Unit 2 (dosimetry analysis & storage) Note 1	Z	8	7/1/93	2.41	WCAP-14231
Unit 2 (dosimetry analysis & storage) Note 1	Y	8	7/1/93	2.08 [b]	WCAP-14231
Ex-vessel Cavity Dosimetry	N/A	12	3/12/99	--	WCAP-15334

a. Approximate fluence at vessel 1/4 thickness location, at 32 EFPY
b. Approximate fluence at vessel inner wall location, at 32 EFPY
c. Approximate fluence at vessel 1/4 thickness location, at 54 EFPY
d. Approximate fluence at vessel inner wall location at 54 EFPY

Note 1 – Capsule specimens have been removed and stored at Westinghouse after reading dosimetry. These specimens are available for testing or additional irradiation if ever deemed necessary.

Note 2 – The management of capsule W is controlled by a sister plant with greater EOL fluence projections than McGuire. Presently they plan to test this capsule after the EOC-18 withdrawal.

For Catawba Unit 2, Capsule U is not necessary. The EOL predicted shift (ΔRT_{NDT}) is less than 100° F, therefore only 3 capsules are required to meet the requirements of ASTM E-185 referenced in 10 CFR 50, Appendix H. Catawba Unit 2 is utilizing 5 capsules for its surveillance program. The treatment of Catawba Unit 2, Capsule U will be clarified by revising the Catawba UFSAR Table 18.2 to read as follows:

Catawba Reactor Vessel Capsule Withdrawal Schedule

Unit	Capsule	End of Cycle (EOC)	Projected EOC Date	Estimated Fluence ($n/cm^2 \times 10^{19}$)	Reference
Unit 1	Z	1	8/8/86	0.299	WCAP -11527
Unit 1	Y	6	7/10/92	1.32[a]	WCAP-13720
Unit 1	W	14	11/29/03	3.0 [d]	--
Unit 1 (dosimetry analysis & storage)	X	10	12/20/97	2.44	WCAP-15117
Unit 1 (dosimetry analysis & storage) Note 1	U	10	12/20/97	2.44	WCAP-15117
Unit 1	V	10	12/20/97	2.33 [b][c]	WCAP-15117
Ex-vessel Cavity Dosimetry	N/A	13	2003 RFO	--	--
Unit 2	Z	1	12/23/87	0.323	WCAP-11941
Unit 2	X	5	1/23/93	1.23[a]	WCAP-13875
Unit 2	W	14	3/9/06	3.0 [d]	--
Unit 2	U	Note 2	Note 2	---	--
Unit 2 (dosimetry analysis & storage) Note 1	Y	9	9/13/98	2.49	WCAP-15243
Unit 2	V	9	9/13/98	2.38 [b][c]	WCAP-15243
Ex-vessel Cavity Dosimetry	N/A	13	2004 RFO	--	--

a. Approximate fluence at vessel 1/4 thickness location, at 32 EFPY
b. Approximate fluence at vessel inner wall location, at 32 EFPY
c. Approximate fluence at vessel 1/4 thickness location, at 54 EFPY
d. Approximate fluence at vessel inner wall location at 54 EFPY

Note 1 – Capsule specimens have been removed and stored at Westinghouse after reading dosimetry. These specimens are available for testing or additional irradiation if ever deemed necessary.

Note 2 – Capsule U is not available for irradiation and testing.

For McGuire Units 1 and 2 and Catawba Units 1 and 2, as stated in the Application (Section B.3.26), all pulled capsules have either been tested or stored. The cavity dosimetry activity within the *Reactor Vessel Integrity Program* has been established at McGuire and will be installed in Catawba Units 1 and 2 in upcoming outages.

The plant specific vessel capsule withdrawal schedules provided in the summary description of the *Reactor Vessel Integrity Program* of each station's UFSAR Supplement will be revised to reflect the changes made in this response.

<u>New open item 3.1.3.2.2-2.</u> The staff and nuclear power industry are pursuing resolution of the reactor vessel penetration nozzle cracking issue associated with the Davis Besse boric acid corrosion and reactor vessel head wastage issue identified in October 2000. The staff is evaluating potential changes to the requirements governing inspections of Alloy 600 vessel head penetration (VHP) nozzles and PWR upper RV heads (specifically with respect to non-destructive examinations and the ability to detect cracking in the VHP nozzles prior to loss of material in the upper RV heads). This is an emerging issue that has not yet been resolved and is beyond the scope of this license renewal review, pursuant to 10 CFR 54.30(b). However, since this issue might not be resolved prior to issuance of the renewed operating licenses for the McGuire and Catawba units, the staff requests the applicant to commit to implementing any actions, as part of the VHP Nozzle Program, that are agreed upon between the NRC, NEI, MRP, and the nuclear power industry to monitor for, detect, evaluate, and correct cracking the VHP nozzles of U.S. PWRs, specifically as the actions relate to ensuring the integrity of VHP nozzles in the McGuire and Catawba upper RV heads during the extended period of operation. This commitment will ensure that the applicant's VHP Nozzle Program (as described in the McGuire and Catawba UFSARs) will be capable of monitoring for, detecting, evaluating, and correcting cracking in the McGuire and Catawba VHP nozzles and associated upper RV heads before unacceptable degradation of the VHP nozzles or associated upper RV heads occurs. Any updates to the VHP Nozzle Program that result from resolution of this issue should be reflected in the McGuire and Catawba UFSARs.

Duke Response to New Open Item 3.1.3.2.2-2
In response to New Open Item 3.1.3.2.2-2, Duke incorporates by reference (pursuant to §54.17(e)) its response to NRC Bulletin 2002-02 dated September 6, 2002. The following regulatory commitments were made by Duke in response to this bulletin:

(1) Catawba and McGuire Nuclear Stations will supplement their Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle inspection programs with non-visual NDE methods.
(2) Plans will be submitted that more specifically address methods, scope, coverage, frequencies, qualification requirements, and acceptance criteria for future Catawba and McGuire inspections of the Reactor Pressure Vessel Head and Vessel Head Penetration Nozzles within four years of the date of this response.

In addition, the *Alloy 600 Aging Management Review* described in Appendix B.3.1 of the Application will be performed to ensure that nickel-based alloy locations are adequately inspected by the *Inservice Inspection Plan* (Appendix B.3.20) or other existing programs such as the *Control Rod Drive Mechanism and Other Vessel ClosurePenetration Program*

(Appendix B.3.9), the *Reactor Vessel Internals Inspection* (Appendix B.3.27), *and the Steam Generator Surveillance Program* (Appendix B.3.31). The review will demonstrate that the general oversight and management of cracking due to primary water stress corrosion cracking (PWSCC) is effective for the period of extended operation.

The summary description of the *Alloy 600 Aging Management Review* contained in each station's UFSAR Supplement will be revised to add the following:

> Consideration of industry operating experience is part of the *Alloy 600 Aging Management Review.* The NRC staff is currently reviewing industry experience with Alloy 600 locations as a result of the Davis-Besse event in March 2002. Any future regulatory actions that may be required as a result of this review will be provided by the staff in separate generic communications to all plants.
>
> The summary aging management program descriptions contained in this UFSAR will be updated as necessary to reflect any new or revised commitments made by Duke in response to the staff generic communication's that result from this event.

The summary description of the *Control Rod Drive Mechanism and Other Vessel Closure Penetration Inspection Program* contained in each station's UFSAR Supplement will be revised to add the following:

> This summary description will be updated as necessary to reflect any new or revised commitments made by Duke in response to the staff generic communication's that result from the Davis-Besse event in March 2002.

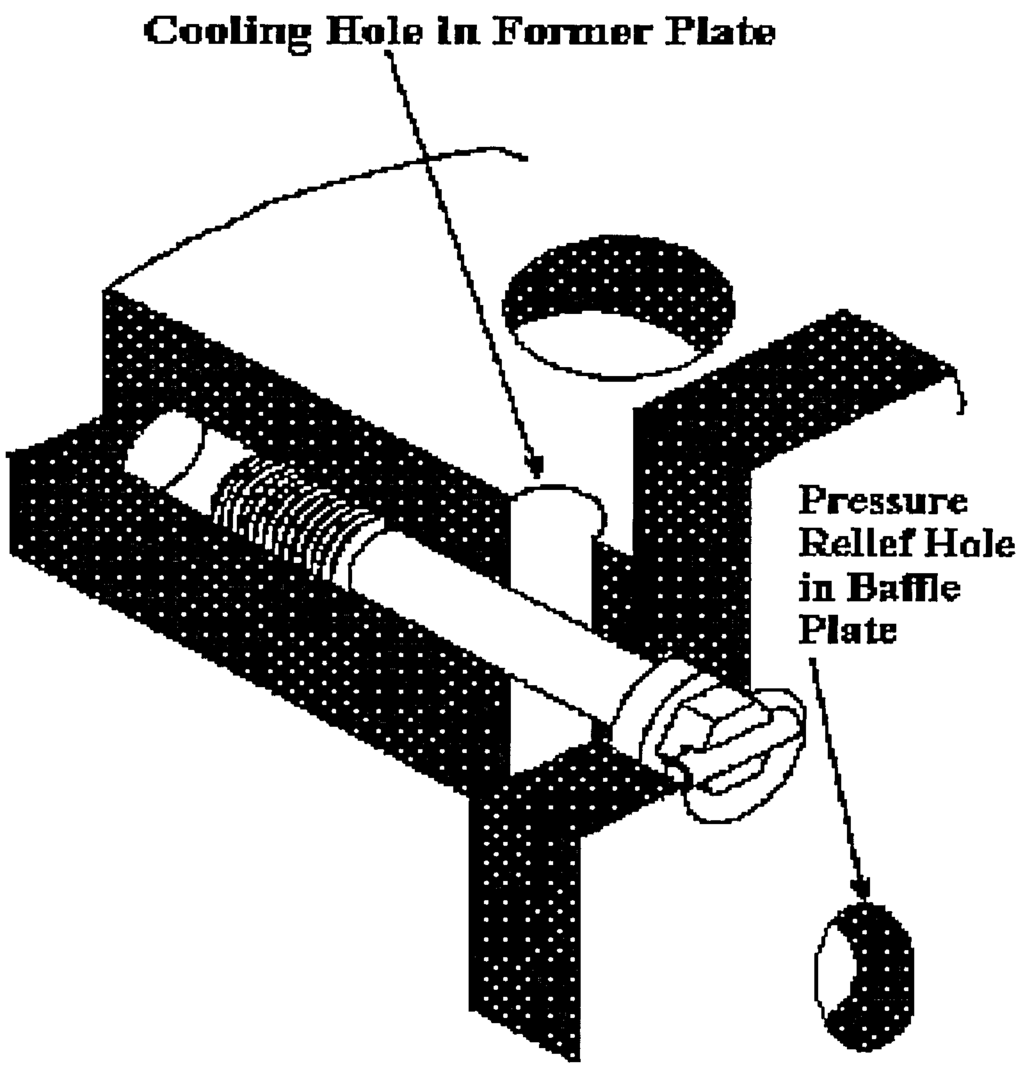
New Open Item 3.1.4-1(a) Since the fabricator for the McGuire 1 and Catawba 2 RVs is not the same as the design fabricators for McGuire 2 and Catawba 1 RVs or for the Oconee RVs, some uncertainty exists whether the inspections of welded RV internals at Oconee 1 and McGuire 1 will be truly representative of the condition of welded RV internals at McGuire 2 and the Catawba units. The staff's position is that the applicant should schedule inspection of remaining RV internal plates, forgings, welds and bolts (i.e., core barrel bolts and thermal shield bolts) at all of the McGuire and Catawba reactor units.

Duke Response to New Open Item 3.1.4-1(a)

New Open Item 3.1.4-1(a) identifies two primary concerns. First, the staff is concerned about the apparent difference in vessel internals fabricators. However, all the McGuire and Catawba reactor vessel internals are manufactured by Westinghouse – not by the reactor vessel manufacturers. McGuire 1 leads McGuire 2 in operating hours and is clearly the lead plant for reactor vessel internals inspection.

Second, the staff is concerned about the lack of similarity between the McGuire and Catawba internals and those of Oconee. In its letter dated April 15, 2002, Duke provided in its response to RAI B.3.27-1 a table of comparing Oconee Unit 1 relevant information to McGuire Unit 1 and Unit 2 and Catawba Unit 1 and Unit 2. This relevant information includes power levels, baffle former and plate material, baffle bolt material, T_{hot} and T_{cold}, and estimated peak fluenace at baffle plate and bolt location and year. Duke believes that this table clearly indicates the similarities of all of these vessel internals.

The Catawba internals will have much less potential for the referenced aging effects, since they are an original upflow design with cooling holes for the baffle bolts and pressure relief holes in the baffle plates. The stresses are also less due to the lower differential pressure across baffle plates from the bypass region. The diagram below indicates the cooling holes in the former plates and the pressure relief holes in the baffle plates in the Catawba internals.


Cooling Hole in Former Plate
Pressure Relief Hole in Baffle Plate

The only significant weld in the McGuire and Catawba reactor vessel internals is the circumferential weld in the core barrel, which has a much lower fluence than the regions of concern. All other welds are used to capture locking devices. The core barrel and thermal shield bolts which are in the Oconee internals are not part of the McGuire and Catawba design.

For the reasons stated above and as discussed during the technical level meeting with the staff on September 17, 2002, Duke agrees that the McGuire reactor vessel internals will be leading indicators for the Catawba internals and commits to inspect both McGuire Unit 1 and McGuire Unit 2 reactor vessel internals during the period of extended operation.

The UFSAR Supplement for McGuire, Reactor Vessel Internals Inspection, **Monitoring & Trending,** 4th paragraph will be revised as follows:

> McGuire Unit 1 will be inspected in the fifth inservice inspection interval. McGuire Unit 2 will be inspected early in the sixth inservice interval (prior to the last year of the 20-year period of extended operation).

The UFSAR Supplement for Catawba, Reactor Vessel Internals Inspection, **Monitoring & Trending,** 4th paragraph will be revised as follows:

> McGuire Unit 1 will be inspected in the fifth inservice inspection interval. McGuire Unit 2 will be inspected early in the sixth inservice interval (prior to the last year of the 20-year period of extended operation). The decision to perform inspections on Catawba Unit 1 and Catawba Unit 2 will depend on an evaluation of the internals inspections performed on McGuire Units 1 and 2.

New Open Item 3.1.4-1(b) The critical crack size acceptance criterion for RV internal forgings, plates, and welds, and RV internals made from CASS have not yet been established. Nor have any acceptance criteria been proposed for the inspections that might be proposed to monitor the RV internals for void swelling. The applicant will need to submit the critical crack size acceptance criteria for the RV internal forgings, plates, and welds, and RV internals made from CASS once the evaluations for these components have been completed and the critical crack sizes for these components have been established. Once the applicant has finalized its evaluation of void swelling of the RV internals, the applicant will also need to submit the acceptance criteria for dimensional changes that might result in the RV internal components as a result of void swelling.

Duke Response to New Open Item 3.1.4-1(b)
In response to New Open Item 3.1.4-1(b), the Acceptance Criteria attribute of the Reactor Vessel Internals Inspection summary description contained in each station's UFSAR Supplement will be revised to read as follows (revision text underlined):

> **Acceptance Criteria** – The *Reactor Vessel Internals Inspection* includes the following acceptance criteria:
>
> For the items comprised of plates, forgings, and welds, critical crack size will be determined by analysis <u>and submitted for review and approval to the NRC staff</u> prior to the inspection.
>
> For baffle bolts, any detectable crack indication is unacceptable for a particular baffle bolt. The number of baffle bolts needed to be intact and their locations will be determined by analysis.
>
> For items fabricated from CASS, critical crack size will be determined by analysis. Acceptance criteria for all aging effects will be developed <u>and submitted for review and approval to the NRC staff</u> prior to the inspection.
>
> <u>For items subject to dimensional changes due to void swelling, activities are in progress to develop and qualify the inspection method. Acceptance criteria will be developed and submitted for review and approval to the NRC staff prior to the inspection.</u>

New Open Item 3.1.4-1(c) The staff requests that Duke provide a commitment to update the "Detection of Aging Effects" program attribute in FSAR Supplement Section 18.2.22, "Reactor Vessel Internals Inspection," to reflect the second paragraph in the applicant's response to RAI B.27-2.

Duke Response to New Open item 3.1.4-1(c)

In response to New Open item 3.1.4-1(c), the following statement will be added to the plates, forgings, and welds visual inspection portion of **Monitoring & Trending** attribute of the *Reactor Vessel Internals Inspection* summary description contained in each station's UFSAR Supplement:

> The visual inspection method selected for the inspection of RV internal plates, forging, and welds will be sufficient to detect cracks in the components prior to any growth to a size that is greater than the critical crack size (critical crack length) for the material.

New Open Item 3.1.5-1 The staff requests the applicant to include a reference to NEI 97-06 in a summary description of the Steam Generator Surveillance Program or in Tables 18-1 of the McGuire and Catawba FSAR Supplements.

Duke Response to New Open Item 3.1.5-1
In response to New Open Item 3.1.5-1, the following changes will be incorporated into the UFSAR Supplements for each station:

(1) In Table 18-1, for the *Steam Generator Surveillance Program*, "18.3" will be added to the entry in the "UFSAR/ITS Location" column.
(2) New Section 18.3 will be added (the References section will become Section 18.4) and the following statement will be included in Section 18.3:

The inspections of the *Steam Generator Surveillance Program* follow the recommendations of NEI 97-06, "Steam Generator Program Guidelines."

New Open Item 4.2-1 (NRC letter dated September 13, 2002) So that the NRC staff can effectively complete its review of your license renewal application, the NRC requests that you provide, as a supplement to your license renewal application, a reassessment of the TLAAs provided in Section 4.2 of your license renewal application to demonstrate compliance with the requirements of 10 CFR 50.61 for the McGuire, Unit 1, RPV, during the period of extended operation. The reassessment should take into account the new information from the fourth Diablo Canyon, Unit 2, surveillance capsule. For tracking purposes, an open item designation of 4.2-1 has been established to resolve this issue.

Duke Response to New Open Item 4.2-1

Duke has performed a reassessement of the TLAAs provided in Section 4.2 of the Application for the McGuire Unit 1 reactor vessel taking into account the new information from the fourth Diablo Canyon, Unit 2 surveillance capsule. The results of this reassessment are as follows:

PRESSURIZED THERMAL SHOCK REASSESSMENT

The information from the fourth Diablo Canyon, Unit 2 surveillance capsule results in changes to the margin term (M) in the two rows of Table 4.2-5 RT PTS Calculations for McGuire Unit 1 Beltline Region Materials at 54 EFPY indicated by **BOLD**:

Material	CF	Fluence @ 54 EFPY(a)	FF	RTNDT(U) (b)	Δ RT_{PTS} (c)	M	RT_{PTS} (d)
Lower Shell Plate Long. Weld Seams 3-442A,C (e) (30° Azimuth)	208.2	2.73	1.27	-50	264.4	56	270
→ Using S/C Data from Diablo Canyon 2	**194.4**	**2.73**	**1.27**	**-50**	**246.9**	**56**	**253**
Lower Shell Plate Long. Weld Seams 3-442B (e) (0° Azimuth)	208.2	1.89	1.17	-50	243.6	56	250
→ Using S/C Data from Diablo Canyon 2	**194.4**	**1.89**	**1.17**	**-50**	**227.4**	**56**	**233**

Notes:

(a) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E > 1.0 MeV).

(b) Information provided in WCAP-15192[13], Table 4-5. Initial RT_{NDT} values are measured values.

(c) ΔRT_{PTS} = CF * FF

(d) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS}$ + Margin (°F)

(e) Heat # 21935/12008

As shown above all of the beltline region materials in the McGuire Units 1 reactor vessel continue to have RT_{PTS} values at 54 EFPY below the screening criteria values of 270°F for plates, forgings and longitudinal welds and 300°F for circumferential welds. Specifically, the lower shell plate longitudinal welds 3-442A & C were the most limiting material for McGuire Unit 1 with a 54 EFPY PTS value of 253°F as compared to the 54 EFPY PTS value of 225°F provided in the Application.

UPPER SHELF ENERGY (USE)

To evaluate the impact of new data to the USE reported in Table 4.2-1 of the Application, Duke applied the chemistry data from the surveillance capsule report, WCAP-15423, concerning the same weld wires Heat 12008 and 21935 and Linde 1092 Flux Lot as McGuire Unit 1 Lower Shell Longitudinal Weld Seams 3-442A, B, C. The percent copper changed from 0.213% (as reported in the Application) to 0.219% (as reported in WCAP-15423). Using Figure 2 of RG 1.99, Rev. 2, the difference in USE is less than a 0.5 % drop. Therefore, the EOL USE would conservatively be 1°F less than the values provided in Table 4.2-1 of the Application and still above the regulatory limit of 50 ft-lb.

PRESSURE-TEMPERATURE (P-T) LIMITS

The effect of this new data to projected Pressure-Temperature limit curves for the period of extended operation as been assessed. Duke has determined that there would be no significant impact on these curves. The conclusion provided in Section 4.2.3 of the Application remain valid.

Table 5-W, RT PTS Calculations for McGuire Unit 1 Beltline Region Materials at 54 EFPY contained in the McGuire UFSAR Supplement will be revised to include the revised values provided in this response.

Open Item 4.3-3 The staff reviewed the Catawba Updated Final Safety Analysis Report (UFSAR), Section 1.7, Regulatory Guides, and Section 5.3.1.4, Special Controls for Ferritic and Austenitic Stainless Steels, and determined that sufficient information was provided in the UFSAR to conclude that underclad cracking was not a concern for Catawba 1 and 2. The staff also reviewed information, submitted by letter from the applicant dated July 9, 2002, to conclude that underclad cracking is not a concern for McGuire 1. However, the staff does not have sufficient information about the McGuire 2 fabrication process to conclude that underclad cracking is not a concern. If the applicant can not provide conclusive evidence that the fabrication procedure does not result in underclad cracking, then it can furnish an analysis for the license renewal term.

Duke Response to Open Item 4.3-3
The issue of reactor vessel nozzle underclad cracking was identified, addressed, and resolved during the initial licensing of both McGuire units. A chronology of events and excerpts from relevant correspondence is described in Duke letter dated July 9, 2002. Copies of the two Westinghouse reports (one for McGuire 2 and one for Catawba 1) referenced in the letter were provided to the staff. These reports are listed as references in WCAP-15338, "A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants."

Each report provides substantial details relative to nozzle cladding procedure (fabrication process), ultrasonic examination procedure, calibration, raw data sheets, and ultrasonic reflector plots. The nozzle cladding fabrication process described in each report is essentially the same – post-weld heat treatment / stress relief time is slightly longer for McGuire Unit 2 than Catawba Unit 1.

Duke believes that these reports provide sufficient information to conclude that reactor vessel nozzle underclad cracking is not an issue today – consistent with the staff's conclusion approximately 20-years ago – and is not an aging effect of concern for license renewal and the period of extended operation.

Duke met with the staff on September 17, 2002 and discussed this Open Item. During this meeting, the staff reiterated its position that Duke need not address this issue for McGuire Unit 1. However, underclad cracking remains a staff concern for McGuire Unit 2 because Duke is continuing to rely upon ultrasonic inspection and the evaluation of indications found during the inspection that clearly indicate that the indications were within the acceptance criteria of ASME Section III for resolution of this issue on this unit. The staff stated its belief that an ultrasonic inspection may not be fully effective at detecting defects of the size generated by this phenomenon. Therefore, this issue can be resolved for McGuire Unit 2 only by analysis.

Duke continues to disagree with the staff position. Nevertheless and as a practical matter in order to support the timely resolution of the open item and the completion of the license renewal review on schedule, Duke will not challenge this issue further.

WCAP-15338, "A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants" contains an analysis that has been used to resolve this issue for McGuire Unit 2. The NRC staff has reviewed this report and documented its findings in a safety evaluation report dated September 25, 2002. While the staff has approved WCAP-15338 generically, two license renewal applicant action items were identified that must be addressed by license renewal applicants when incorporating WCAP-15338 into a license renewal application.

Renewal Applicant Action Item (1) states:

> The license renewal applicant is to verify that its plant is bounded by the WCAP-15338 report. Specifically, the renewal applicant is to indicate whether the number of design cycles and transients assumed in WCAP-15338 analysis bounds the number of cycles for 60 years of operation of its RPV.

Duke Specific Response to Renewal Applicant Action Item (1) for McGuire Unit 2:

> Duke has compared the number of design cycles and transients used in the analysis contained in WCAP-15338 with the applicable number of design cycles and transients contained in McGuire Unit 2 design documents and verifies that WCAP-15338 bounds McGuire Unit 2. Duke notes that Catawba 1 is also bounded by WCAP-15338.

Renewal Applicant Action Item (2) states:

> Section 54.21(d) of 10 CFR requires that an FSAR supplement for the facility contains a summary description of the programs and activities for managing the effects of aging and the evaluation of TLAA for the period of extended operation. Those applicants for license renewal referencing WCAP-15338 report for the RPV components shall ensure that the evaluation of the TLAA is summarily described in the FSAR supplement.

Duke Specific Response to Renewal Applicant Action Item (1) for McGuire Unit 2:

The McGuire UFSAR Supplement will be revised to include the following:

Reactor Vessel Nozzle Underclad Cracking
The staff identified during its review of the license renewal application the issue of reactor vessel nozzle underclad cracking. A summary of the history of this issue was provided in Duke letter dated July 9, 2002 to the NRC wherein Duke concluded that the issue was not a time-limited aging analysis for license renewal. NRC letter to Duke dated August 14, 2002 stated that this is a license renewal issue that is applicable only to McGuire Unit 2. Duke provided an analysis of this license renewal issue by demonstrating that a generic and bounding analysis performed by Westinghouse and documented in WCAP-15338, "A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants" was applicable to McGuire Unit 2. Duke performed a review and verified that the analysis contained in WCAP-15338 bounds McGuire Unit 2. Accordingly, the reactor vessel nozzle underclad cracking issue for McGuire Unit 2 was resolved by the staff in its SER for License Renewal (insert date).

Attachment 2

Response to
McGuire Units 1 & 2 and Catawba Units 1 & 2
Safety Evaluation Report with Open Items

Mechanical Related Items

Open Items 2.3-1 and 2.3-2 The applicant failed to perform an AMR for the housings of active components (e.g., fans and dampers) that may perform critical pressure retention and/or structural integrity functions. Failure to maintain that function could prevent the associated active component from performing its function. Since these housings are within the scope of license renewal and are long-lived and passive, they are subject to an AMR in accordance with 10 CFR 54.21.

Duke Response to Open Items 2.3-1 and 2.3-2
Duke disagrees with the staff for the following reasons.

10 CFR 54.21(a)(1) notes that damper and fans without exception are excluded from an aging management review. NEI 95-10, *NEI 95-10 (Revision 2) Industry Guideline for Implementing the Requirements of 10 CFR 54 – The License Renewal Rule,* that is endorsed by the staff as an acceptable method for implementing the requirements of 10 CFR 54 notes in Appendix B that dampers and fans are not passive, and therefore, they are not subject to an aging management review.

NUREG-1800, *Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants,* notes in Table 2.1-5 that dampers and fans are not passive, and therefore, they are not subject to an aging management review. NUREG-1801, *Generic Aging Lessons Learned (GALL) Report,* does not contain any entries for dampers and fans.

The above documents all show that dampers and fans are not subject to an aging management review. In preparing the technical work and the Application, Duke followed the industry and staff guidance documents in effect at the time. As a result, Duke determined that the dampers and fans were within the scope of license renewal but not subject to an aging management review. This is the same position taken during the renewal of the Oconee license that was determined acceptable by the Oconee SER presented in NUREG 1723.

The staff has provided a draft "Staff Position on Screening of Housings for Active Housings" by letter dated May 1, 2002. Currently, this staff position has not been finalized. As the staff stated during a meeting with Duke on September 18, 2002, its interpretation of 10 CFR 54.21 is that fan housings and damper housings perform a pressure boundary function and therefore are subject to aging management review. Duke notes that the process for documenting an interpretation of Part 54 is provided in §54.7, Interpretations. A written staff position would need to be in accordance with this section to be recognized to be binding upon the Commission.

Nevertheless and as a practical matter in order to support the timely resolution of these open items and the completion of the license renewal review on schedule, Duke will not challenge this issue further. The following are the results of the aging management review for the in scope fan housing and damper housings at McGuire and Catawba.

Aging Management Review Results for Fans and Dampers

1	2	3	4	5	6
Component Type	**Component Function**	**Material**	**Internal Environment (Note 1)** **External Environment (Note 2)**	**Aging Effect**	**Aging Management Program and Activity (Note 3)**
Dampers	Pressure Boundary	Aluminum	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Dampers	Pressure Boundary	Carbon Steel	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components Fluid Leak Management Program
Dampers	Pressure Boundary	Galvanized Steel	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program
Dampers	Pressure Boundary	Stainless Steel	Ventilation	None Identified	None Required
			Reactor Building	None Identified	None Required
Fans	Pressure Boundary	Aluminum	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Fans	Pressure Boundary	Carbon Steel	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components Fluid Leak Management Program
Fans	Pressure Boundary	Galvanized Steel	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	None Required

Based on the evaluations provided in Appendix B of the Application for the aging management programs and activities listed above, the aging effects will be adequately managed such that the intended functions of the fans and dampers will be maintained consistent with the current licensing basis for the period of extended operation. No revisions to the UFSAR Supplement summary descriptions of these programs are required.

Notes for Aging Management Review Results for Fans and Dampers:

Pressure Boundary = Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered

(1) **Internal Environment**
Ventilation = Ambient air that is conditioned to maintain a suitable environment for equipment operation and personnel occupancy

(2) **External Environment**
Aluminum fans and dampers are located in the Standby Shutdown Facility.
Galvanized steel fans are located in the Standby Shutdown Facility.

(3) **Aging Management Programs and Activities**
Fluid Leak Management Program applies only to components in Auxiliary Building.
Inspection Program for Civil Engineering Structures and Components applies in all locations.

Open Item 2.3-3

The AMP (the Inspection Program for Civil Engineering Structures and Components) credited by the applicant for monitoring the aging of structures that include structural sealants as sub-components does not include, within its scope, building sealants. Therefore, this AMP is not adequate to manage the aging of building sealants, which are long-lived, passive structural sub-components within the scope of license renewal.

Duke Response to Open Item 2.3-3

In response to Open Item 2.3-3, Duke would like to summarize its previous responses to this staff concern.

As stated in our response to RAI 2.3-4, Duke does not define materials such as ventilation area pressure boundary sealants as structures or components. The guidance provided in NUREG-1800, "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants," states that structural sealants are "considered as subcomponents and are not explicitly called out in the scoping and screening procedures." Furthermore, the Commission in the SOC for the Final Part 54 Rule stated:

> "... the Commission has removed the words "portions of" and similar wording from the Statements of Consideration when it could be misinterpreted to mean a subcomponent piece-part demonstration."

Aging management reviews are required for structures and components – not subcomponents. Although ventilation area pressure boundary sealants are not listed as components in the LRA, Duke will assume that pressure boundary they structural sealants are subject to aging management review. Pressure boundary structural sealants include, but are not limited to, sealants in the interface between a structural wall, floor or ceiling and a non-structural component such as duct, piping, electrical cables, doors, and non-structural walls.

The function supported by these structural sealants is to minimize inleakage of building pressure boundary enclosure and to maintain a differential pressure between the ventilation area and the adjacent structural areas. In some instances, the amount of assumed inleakage is quantified. In other instances, the design basis simply states that inleakage should be minimized. The structural sealants are located in benign environments and may not be susceptible to significant degradation resulting in loss of function. However, for the purpose of this review, the aging effects of concern are assumed to be cracking and shrinkage of the structural sealants.

Duke has previously proposed crediting existing technical specification surveillances that provide assurance that the design basis functions are being met. All of these surveillances, except the Control Room surveillance which is the subject of an ongoing regulatory issue, verify that the function of the ventilation area boundary, including the structural sealants, is being managed:

- The sealants for the Control Room pressure boundary enclosure are addressed by surveillance testing to demonstrate compliance with *McGuire Technical Specification 3.7.9 and Catawba Technical Specification 3.7.10.*
- The sealants for the Auxiliary Building (VA) ventilation pressure boundary enclosure are addressed by surveillance testing to demonstrate compliance with *McGuire Technical Specification 3.7.11.4 and Catawba Technical Specification 3.7.12.4.*
- The sealants for the Fuel Building (VF) ventilation pressure boundary enclosure are addressed by surveillance testing to demonstrate compliance with *McGuire Technical Specification 3.7.12.4 and Catawba Technical Specification 3.7.13.4.*
- The sealants for the Reactor Building (annulus) (VE) ventilation pressure boundary enclosure are addressed by surveillance testing to demonstrate compliance with *McGuire and Catawba Technical Specification 3.6.10.5.*

During a meeting with the staff on September 18, 2002, the staff indicated that these surveillances do not directly manage the aging of the structural sealants. Duke agrees that the aging of the structural sealants are not directly inspected. However, the function of the sealants and the ventilation area pressure boundary, except those of the control room boundary, are being managed with reasonable assurance by the specified surveillance programs. In the event the acceptance criteria of the surveillances are not met, the entire ventilation area pressure boundary will be inspected to determine the cause of the excess inleakage. Corrective actions are taken to repair or replace the ineffective sealant.

Nevertheless and as a practical matter in order to support the timely resolution of this open item and the completion of the license renewal review on schedule, Duke will not challenge this issue further. Duke will implement a *Ventilation Area Pressure Boundary Sealants Inspection* to manage these sealants. The following is a description of this new one-time inspection. Duke proposes to enhance existing surveillance requirements for the period of extended operation by implementing visual inspections of the structural sealants that function to establish the ventilation pressure boundary of the Control Room, ECCS Pump Room, Annulus, and Fuel Handling areas. The following is the description of this new inspection activity.

VENTILATION AREA PRESSURE BOUNDARY SEALANTS INSPECTION
The purpose of the *Ventilation Area Pressure Boundary Sealants Inspection* is to enhance existing surveillance requirements characterize any cracking or shrinkage of structural sealants due to exposure to the ambient conditions. Uncertainty exists as to whether exposure of pressure boundary structural sealants to the ambient conditions within the Auxiliary Building, Annulus and Fuel Handling Building could cause cracking or shrinkage and result in a loss of function of the sealants. to provide additional assurance that the structural sealants installed in the ventilation pressure boundary of the Control Room, ECCS Pump Room, Annulus, and Fuel Handling areas will continue to maintain the differential pressure required by the current licensing basis. The visual inspection will identify cracking and shrinkage of the structural sealants that would result in loss of intended function and an inability of the sealants to maintain the differential pressure required by the current design basis. Corrective actions may then be taken to repair or replace the structural sealants. The *Ventilation Area Pressure Boundary Sealants Inspection* is a one-time inspection.

Scope – The scope of the *Ventilation Area Pressure Boundary Sealants Inspection* is the pressure boundary structural sealants installed in the ventilation pressure boundary of the Control Room, ECCS Pump Room, Annulus, and Fuel Handling areas. Pressure boundary structural sealants include, but are not limited to, sealants in the interface between a structural wall, floor or ceiling and a non-structural component such as duct, piping, electrical cables, doors, and non-structural walls.

Preventive Actions – No actions are taken as a part of this surveillance one-time inspection to prevent aging effects or to mitigate aging degradation.

Parameters Monitored or Inspected – *Ventilation Area Pressure Boundary Sealants Inspection* is a visual inspection for cracking or shrinkage of the structural sealants.

Detection of Aging Effects – In accordance with the information provided in **Monitoring & Trending**, *Ventilation Area Pressure Boundary Sealants Inspection* will detect cracking or shrinkage of the ventilation area pressure boundary structural sealants.

Monitoring & Trending – The *Ventilation Area Pressure Boundary Sealants Inspection* will will visually inspect a representative sample of structural sealants at each station. Locations of inspections will be based on severity of the local ambient conditions taking into consideration temperature and radiation. The sample locations selected will provide a leading indication of the condition of all structural sealants within the scope of this activity.

No actions are taken as part of this program to trend inspection results.

For McGuire, this new one-time inspection will be completed following issuance of the renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1)

For Catawba, this new one-time inspection will be completed following issuance of the renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba Unit 1).

Acceptance Criteria – The acceptance criterion for the *Ventilation Area Pressure Boundary Sealants Inspection* is no unacceptable cracking or shrinking that could result in the loss of the intended function of the structural sealant as determined by engineering evaluation.

Corrective Action & Confirmation Process – If engineering evaluation determines that continuation of the aging effects will not cause a loss of structural sealant intended function, under any current licensing basis design condition for the period of extended operation, no further action is required. If the engineering evaluation determines that continuation of the aging effects could cause a loss of structural sealant function under current licensing design conditions for the period of extended operation, then programmatic oversight will be defined by engineering. Specific corrective actions, including repair or replacement of the ventilation area pressure boundary structural sealants, will be are implemented in accordance with the corrective action program.

Administrative Controls – *Ventilation Area Pressure Boundary Sealants Inspection* surveillances will be implemented by written procedure.

Operating Experience – The *Ventilation Area Pressure Boundary Sealants Inspection* is a new one-time inspection activity for which there is no operating experience. However, similar visual inspections have been performed as part of the *Inspection Program for Civil Engineering Structures and Components* which has been found to be an acceptable aging management program for license renewal by the staff.

UFSAR SUPPLEMENT REVISIONS

Each station's UFSAR Supplement will be revised to include the following description of the *Ventilation Area Pressure Boundary Sealants Inspection*:

Scope – The scope of the *Ventilation Area Pressure Boundary Sealants Inspection* is the pressure boundary structural sealants installed in the ventilation pressure boundary of the Control Room, ECCS Pump Room, Annulus, and Fuel Handling areas. Pressure boundary structural sealants include, but are not limited to, sealants in the interface between a structural wall, floor or ceiling and a non-structural component such as duct, piping, electrical cables, doors, and non-structural walls.

Preventive Actions – No actions are taken as a part of this one-time inspection to prevent aging effects or to mitigate aging degradation.

Parameters Monitored or Inspected – *Ventilation Area Pressure Boundary Sealants Inspection* is a visual inspection for cracking or shrinkage of the structural sealants.

Detection of Aging Effects – In accordance with the information provided in **Monitoring & Trending**, *Ventilation Area Pressure Boundary Sealants Inspection* will detect cracking or shrinkage of the ventilation area pressure boundary structural sealants.

Monitoring & Trending – The *Ventilation Area Pressure Boundary Sealants Inspection* will visually inspect a representative sample of structural sealants at each station. Locations of inspections will be based on severity of the local ambient conditions taking into consideration temperature and radiation. The sample locations selected will provide a leading indication of the condition of all structural sealants within the scope of this activity.

No actions are taken as part of this program to trend inspection results.

For McGuire, this one-time inspection will be completed following issuance of the renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1)

For Catawba, this one-time inspection will be completed following issuance of the renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba Unit 1).

Acceptance Criteria – The acceptance criterion for the *Ventilation Area Pressure Boundary Sealants Inspection* is no unacceptable cracking or shrinking that could result in the loss of the intended function of the structural sealant as determined by engineering evaluation.

Corrective Action & Confirmation Process – If engineering evaluation determines that continuation of the aging effects will not cause a loss of structural sealant intended function, under any current licensing basis design condition for the period of extended operation, no further action is required. If the engineering evaluation determines that continuation of the aging effects could cause a loss of structural sealant function under current licensing design conditions for the period of extended operation, then programmatic oversight will be defined by engineering. Specific corrective actions, including repair or replacement of the ventilation area pressure boundary structural sealants, will be implemented in accordance with the corrective action program.

Administrative Controls – *Ventilation Area Pressure Boundary Sealants Inspection* surveillances will be implemented by written procedure.

Open Item 2.3.3.12.2-1 By letter dated January 28, 2002, the staff requested, in RAI 2.3.3.12-1, that the applicant provide the basis for not listing the turbocharger turbine flexible hose in Table 3.3-15, since these components are passive, long-lived, and have pressure boundary intended functions. In its response, dated April 15, 2002, the applicant stated that the flexible hose is replaced during periodic maintenance. The applicant implied that the hose is replaced based on qualified life in accordance with 10 CFR 54.21(a)(1)(i) and is, therefore, not subject to an AMR. However, since this was not clearly stated in the RAI response, this issue is characterized as an open item.

Duke Response to Open Item 2.3.3.12.2-1
The flexible hose in the Diesel Generator Cooling Water System is replaced on a qualified life. The flexible hose is replaced on a six-year frequency. Therefore, the flexible hose is not subject to an aging management review.

Open Item 2.3.3.13.2-1 The applicant did not provide sufficient information in its response to RAI 2.3.3.13-1 to enable the staff to evaluate the adequacy of its replacement of synthetic rubber flexible expansion joints associated with the emergency diesel generator crankcase vacuum system during periodic maintenance. The applicant should indicate if replacement of these components is based upon a qualified life or based upon condition or performance monitoring. If replacement is based upon the latter, the applicant should specify the parameters that will be monitored as indicators of the components' condition or performance.

Duke Response to Open Item 2.3.3.13.2-1
The flexible hose on the inlet and outlet the of the diesel generator crankcase vacuum blowers are replaced based on condition. The synthetic rubber flexible hose is inspected for cracking and signs of wear on a six-year frequency.

Open Item 2.3.3.14.2-1 The applicant did not provide sufficient information in its response to RAI 2.3.3.14-1 to enable the staff to evaluate the adequacy of its replacement of flexible hose connections associated with the emergency diesel generator fuel oil system during periodic maintenance. The applicant should indicate if replacement of these components is based upon a qualified life or based upon condition or performance monitoring. If replacement is based upon the latter, the applicant should specify the parameters that will be monitored as indicators of the components' condition or performance.

Duke Response to Open Item 2.3.3.14.2-1
The flexible hoses in the Diesel Generator Fuel Oil System are replaced on a qualified life. The flexible hoses are replaced on a six-year frequency. Therefore, the flexible hoses are not subject to an aging management review.

Open Item 2.3.3.19-1 McGuire UFSAR Section 9.5.1.2.1 states that fire hydrants are connected to the yard main. Furthermore, fire hydrants are considered passive and long-lived components in accordance with 10 CFR 54.21. Since the UFSAR is referenced in the license conditions for both McGuire and Catawba, and these components are discussed therein as providing a fire suppression function (which is required by 10 CFR 50.48), it appears that these components are required to meet the requirements of 10 CFR 50.48. The UFSAR does not distinguish between those fire hydrants that are required by 10 CFR 50.48 and those that are not. McGuire is required to meet Appendix A to BTP 9.5-1 and Catawba is required to meet the position documented in CMEB 9.5-1. Accordingly, both documents state that "outside manual hose installation should be sufficient to reach any location with an effective hose stream. To accomplish this, hydrants should be installed approximately every 250 feet on the yard main system." Therefore, the applicant should furnish documentation that demonstrates that the excluded fire hydrants are not required by 10 CFR 50.48 or identify these hydrants as being within the scope of license renewal and subject to an AMR.

Duke Response to Open Item 2.3.3.19-1
The hydrants that are within the scope of license renewal are (1) two hydrants at Catawba that were recently installed to mitigate fires at the Nuclear Service Water Pump Structure (see response to RAI 2.3.3.19-10 submitted to the staff on April 15, 2002) and (2) those hydrants connected to the yard main that are not isolable from the flowpath between the main fire pumps and Auxiliary and Reactor Buildings at McGuire and Catawba.

The staff is correct in identifying that the UFSAR does not differentiate between those hydrants that are required for compliance with 10 CFR 50.48 and those that are not. Therefore, review of the UFSAR alone cannot determine those hydrants required to comply with 10 CFR 50.48.

In a meeting between the staff and Duke on October 1, 2002, the staff indicated that two concerns exist with the out-of-scope hydrants. (1) The hydrants may be relied on as a backup suppression supply for areas in the Auxiliary Building of each station that house safety-related and safe shutdown equipment and as such provide a defense-in-depth aspect of the fire protection program to ensure that the plant can be safely shut down or radioactive releases minimized in the event of a fire. (2) The hydrants may be relied on as suppression for a fire in the yard where radioactive releases could be released to the environment. (This second concern is also stated in the staff's SER with open items on page 2-116.)

In response to the first concern, the fire protection system in the Auxiliary Buildings at McGuire and Catawba consists of two headers that feed the automatic and manual suppression systems.

These headers provide sectional isolation capability between the automatic and manual suppression systems such that a single failure cannot cause loss of water supply to both the automatic and manual means of suppression in a given area. As such, defense in depth exists in the fire protection system design in the Auxiliary Building for each station. The hydrants are not relied on to provide programmatic defense in depth for areas in the Auxiliary Building that house safety-related and safe shutdown equipment.

In response to the second concern, no potential sources of radioactive releases are protected against fire by the out-of-scope hydrants at McGuire or Catawba. Radioactive sources in the Reactor Building and Auxiliary Building are separated from the yard by a 3-hour fire barrier.

A review of each station's Fire Protection Review indicates that hydrants are not relied on in the licensing basis to mitigate fires in areas containing safety related or safe shutdown equipment, with the exception of the Catawba Nuclear Service Water Pumphouse hydrants that are included in scope. Each station's Fire Protection Review consists of submittals made to the NRC that includes the response to Appendix A to Branch Technical Position (BTP)APCSB 9.5-1 and the Fire Hazards Analysis. The Fire Protection Review information is kept current by inclusion in each station's Design Basis Document for Fire Protection. The information in the current McGuire and Catawba Fire Protection Design Basis Document is the same as the original response to the BTP and the Fire Hazards Analysis, indicating there have been no program changes with respect to this issue.

In conclusion, based on a review of the Fire Protection Program licensing information and technical review of plant design, Duke concludes that fire hydrants other than those identified above and indicated on drawings submitted to the staff with the Application are not within the scope of license renewal.

Open Item 2.3.3.19-2 Operating license conditions for McGuire and Catawba, Supplement 2 of the McGuire and Catawba Safety Evaluation Reports (SERs) for original licensing, and Section 9.5.1.2.1 of the McGuire and Catawba UFSARs indicate that jockey pumps are provided to prevent frequent starting of the fire pumps by maintaining pressure in the yard mains in accordance with Section 6.b of BTP CMEB 9.5-1 and NFPA 20. The staff is concerned that the applicant has misapplied the QA Condition 3 designation for license renewal scoping purposes and excluded jockey pumps from the scope of license renewal, although the licensing basis of the plants indicates that these jockey pumps are relied upon to perform a function required by 10 CFR 50.48.

Duke Response to Open Item 2.3.3.19-2
The jockey pumps are part of the current licensing basis of McGuire and Catawba in that they exist as a commitment to satisfy the provision of Appendix A to BTP 9.5-1 for McGuire and Appendix A to CMEB 9.5-1 for Catawba. For license renewal, the jockey pumps do not meet the criteria of 10 CFR 54.4(a)(3) because they are not relied on in a safety analysis or plant evaluation to perform a function to demonstrate compliance with 10 CFR 50.48. The success criteria of §50.48 is clearly the ability to safely shut down the plant and minimize radiation releases in the event of a design basis fire. Plant evaluations demonstrate that the plant can be safely shut down in the event of a design basis fire without the function of the jockey pumps.

Nevertheless and as a practical matter in order to support the timely resolution of this open item and the completion of the license renewal review on schedule, Duke will not challenge this issue further. This response provides the aging management review results for the pressure maintenance subsystem of the fire protection system containing the jockey pump. This subsystem includes an accumulator tank that maintains the pressure on the main fire header during normal plant operation, the jockey pumps that refill the accumulator tank, a source of water for the system refill, and the piping and other in-line components contained in the subsystem. Figures of this portion of the fire protection system are contained in Chapter 9 of each station's UFSAR. For convenience of the reader, larger drawings have been provided separately to the license renewal project manager.

McGuire

Section 3.3 of the Application is supplemented with the following table entries. The table entries below contain components of the McGuire Fire Protection Systems (Interior and Exterior) and the McGuire Condenser Circulating Water System. Some entries already exist in Table 3.3-8 and Table 3.3-26 of the Application but are repeated here for convenience of the reviewer. The following table contains all of the component types of the fire protection pressure maintenance subsystem. The information contained in the table was obtained in the manner described in Section 3.3.1 of the Application.

McGuire Nuclear Station – Aging Management Review Results
Fire Protection Pressure Maintenance Subsystem

1	2	3	4	5	6
Component Type	**Component Function**	**Material**	**Internal Environment** / **External Environment**	**Aging Effect**	**Aging Management Program**
Exterior / Interior Fire Protection System					
Pipe	Pressure Boundary	Carbon Steel	Raw Water	Loss of Material	Service Water Piping Corrosion Program Galvanic Susceptibility Inspection
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Pipe	Pressure Boundary	Ductile iron	Raw Water	Loss of Material	Service Water Piping Corrosion Program Galvanic Susceptibility Inspection
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Pipe	Pressure Boundary	Ductile iron	Raw Water	Loss of Material	Service Water Piping Corrosion Program Galvanic Susceptibility Inspection
			Underground	Loss of Material	Preventive Maintenance Activities-Condenser Circulating Water System Internal Coating Inspection
Pipe	Pressure Boundary	Galvanized Steel	Air / Gas	Loss of Material	Fire Protection Program - Tank and Connected Piping Internal Inspection
			Sheltered	None Identified	None Required

1	2	3	4	5	6
Component Type	**Component Function**	**Material**	**Internal Environment** / **External Environment**	**Aging Effect**	**Aging Management Program**
Pipe	Pressure Boundary	Galvanized Steel	Raw Water	Loss of Material	Service Water Piping Corrosion Program Galvanic Susceptibility Inspection
			Sheltered	None Identified	None Required
Pump Casings (FP Jockey Pumps)	Pressure Boundary	Cast Iron	Raw Water	Loss of Material	Service Water Piping Corrosion Program Selective Leaching Inspection
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Jockey Pump Strainer Housing	Pressure Boundary	Cast Iron	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Jockey Pump Strainer Basket	Filtration	Stainless Steel	Raw Water	Loss of Material	Fire Protection Program – Jockey Pump Strainer Inspection
			No External Environment	Not Applicable	Not Applicable
Tank (FP Accumulator Tank)	Pressure Boundary	Carbon Steel	Air / Gas	Loss of Material	Fire Protection Program - Tank and Connected Piping Internal Inspection
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components

1	2	3	4	5	6
Component Type	Component Function	Material	Internal Environment / External Environment	Aging Effect	Aging Management Program
Tank (FP Accumulator Tank)	Pressure Boundary	Carbon Steel	Raw Water	Loss of Material	Fire Protection Program - Tank and Connected Piping Internal Inspection
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Valve Bodies	Pressure Boundary	Bronze	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Sheltered	None Identified	None Required
Valve Bodies	Pressure Boundary	Cast Iron	Air / Gas	Loss of Material	Fire Protection Program - Tank and Connected Piping Internal Inspection
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Valve Bodies	Pressure Boundary	Cast Iron	Raw Water	Loss of Material	Service Water Piping Corrosion Program Selective Leaching Inspection
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components

1	2	3	4	5	6
Component Type	**Component Function**	**Material**	**Internal Environment** / **External Environment**	**Aging Effect**	**Aging Management Program**
Valve Bodies	Pressure Boundary	Cast Iron	Raw Water	Loss of Material	Service Water Piping Corrosion Program Selective Leaching Inspection
			Underground	Loss of Material	Preventive Maintenance Activities - Condenser Circulating Water System Internal Coating Inspection
Valve Bodies	Pressure Boundary	Carbon Steel	Air / Gas	Loss of Material	Fire Protection Program - Tank and Connected Piping Internal Inspection
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Valve Bodies	Pressure Boundary	Carbon Steel	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Condenser Circulating Water System					
Pipe	Pressure Boundary	Carbon Steel	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Valve Bodies	Pressure Boundary	Carbon Steel	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Based on the evaluations provided in Appendix B for the aging management programs and activities listed above, with the addition of the new activities described below and the additions to the **Scope** attributes for the select programs described below, the aging effects will be adequately managed such that the intended functions of the components listed in the above table will be maintained consistent with the current licensing basis for the period of extended operation.

The following information will be added to the **Scope** attribute of summary description of the *Service Water Piping Corrosion Program* in the McGuire UFSAR Supplement as follows:

- Condenser Circulating Water System

The *Fire Protection Program – Jockey Pump Strainer Inspection* and *Fire Protection Program – Tank and Connected Piping Internal Inspection* are newly identified activities. A ten-attribute activity description and UFSAR Supplement revision are provided for each activity below.

<u>Catawba</u>
Section 3.3 of the Application is supplemented with the following table entries. The table entries below contain components of the Catawba Fire Protection Systems (Interior and Exterior) and the Catawba Filtered Water System. Some entries already exist in Table 3.3-27 of the Application but are repeated here for convenience of the reviewer. The following table contains all of the component types of the fire protection pressure maintenance subsystem. The information contained in the table was obtained in the manner described in Section 3.3.1 of the Application.

Catawba Nuclear Station – Aging Management Review Results
Fire Protection Pressure Maintenance Subsystem

1	2	3	4	5	6
Component Type	**Component Function**	**Material**	**Internal Environment** / **External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Exterior / Interior Fire Protection System					
Pipe	Pressure Boundary	Ductile Iron	Raw Water	Loss of Material	Service Water Piping Corrosion Program Galvanic Susceptibility Inspection
			Underground	Loss of Material	Preventive Maintenance Activities-Condenser Circulating Water System Internal Coating Inspection
Pipe	Pressure Boundary	Ductile Iron	Raw Water	Loss of Material	Service Water Piping Corrosion Program Galvanic Susceptibility Inspection
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Pipe	Pressure Boundary	Galvanized Steel	Air / Gas	None Identified	None Required
			Sheltered	None Identified	None Required
Pipe	Pressure Boundary	Galvanized Steel	Raw Water	Loss of Material	Service Water Piping Corrosion Program Galvanic Susceptibility Inspection
			Sheltered	None Identified	None Required

1	2	3	4	5	6
Component Type	**Component Function**	**Material**	**Internal Environment** **External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Pipe	Pressure Boundary	Stainless Steel	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Sheltered	None Identified	None Required
Pump Casings (FP Jockey Pumps	Pressure Boundary	Cast Iron	Raw Water	Loss of Material	Service Water Piping Corrosion Program Galvanic Susceptibility Inspection Selective Leaching Inspection
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Jockey Pump Strainer Housing (A &B)	Pressure Boundary	Cast Iron	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Jockey Pump Strainer Basket (A & B)	Filtration	Stainless Steel	Raw Water	Loss of Material	Fire Protection Program – Jockey Pump Strainer Inspection
			No External Environment	Not Applicable	Not Applicable
Strainer Housing (C)	Pressure Boundary	Galvanized Steel	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Sheltered	None Identified	None Required
Strainer Basket (C)	Filtration	Stainless Steel	Raw Water	Loss of Material	Fire Protection Program – Jockey Pump Strainer Inspection
			No External Environment	Not Applicable	Not Applicable

1	2	3	4	5	6
Component Type	**Component Function**	**Material**	**Internal Environment** / **External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Tank (FP Accumulator Tank)	Pressure Boundary	Carbon Steel	Air / Gas	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Tank (FP Accumulator Tank)	Pressure Boundary	Carbon Steel	Raw Water	Loss of Material	Fire Protection Program - Tank and Connected Piping Internal Inspection
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Valve Bodies	Pressure Boundary	Brass	Air / Gas	None Identified	None Required
			Sheltered	None Identified	None Required
Valve Bodies	Pressure Boundary	Bronze	Raw Water	Loss of Material	Service Water Piping Corrosion Program Galvanic Susceptibility Inspection
			Sheltered	None Identified	None Required

1	2	3	4	5	6
Component Type	**Component Function**	**Material**	**Internal Environment** / **External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Valve Bodies	Pressure Boundary	Cast Iron	Raw Water	Loss of Material	Service Water Piping Corrosion Program Galvanic Susceptibility Inspection Selective Leaching Inspection
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Valve Bodies	Pressure Boundary	Cast Iron	Raw Water	Loss of Material	Service Water Piping Corrosion Program Galvanic Susceptibility Inspection Selective Leaching Inspection
			Underground	Loss of Material	Preventive Maintenance Activities-Condenser Circulating Water System Internal Coating Inspection
Valve Bodies	Pressure Boundary	Carbon Steel	Raw Water	Loss of Material	Service Water Piping Corrosion Program Galvanic Susceptibility Inspection
			Sheltered	Loss Of Material	Inspection Program for Civil Engineering Structures and Components
Valve Bodies	Pressure Boundary	Stainless Steel	Air / Gas	None Identified	None Required
			Sheltered	None Identified	None Required
Valve Bodies	Pressure Boundary	Stainless Steel	Raw Water	Loss Of Material	Service Water Piping Corrosion Program
			Sheltered	None Identified	None Required

1	2	3	4	5	6
Component Type	**Component Function**	**Material**	**Internal Environment** / **External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Filtered Water System					
Pipe	Pressure Boundary	Aluminum	Raw Water	Loss of Material	Fire Protection Program - Tank and Connected Piping Internal Inspection
			Yard	None Identified	None Required
Pipe	Pressure Boundary	Aluminum	Raw Water	Loss Of Material	Fire Protection Program - Tank and Connected Piping Internal Inspection
			Sheltered	None Identified	None Required
Pipe	Pressure Boundary	Carbon Steel	Raw Water	Loss of Material	Service Water Piping Corrosion Program Galvanic Susceptibility Inspection
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Pipe	Pressure Boundary	Stainless Steel	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Sheltered	None Identified	None Required
Tank (Filtered Water Tank)	Pressure Boundary	Carbon Steel	Raw Water	Loss of Material	Fire Protection Program - Tank and Connected Piping Internal Inspection
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Tank (Filtered Water Tank)	Pressure Boundary	Carbon Steel	Ventilation	Loss of Material	Fire Protection Program - Tank and Connected Piping Internal Inspection
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components

1	2	3	4	5	6
Component Type	**Component Function**	**Material**	**Internal Environment** / **External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Valve Bodies	Pressure Boundary	Aluminum	Raw Water	Loss of Material	Fire Protection Program - Tank and Connected Piping Internal Inspection
			Sheltered	None Identified	None Required
Valve Bodies	Pressure Boundary	Aluminum	Raw Water	Loss of Material	Fire Protection Program - Tank and Connected Piping Internal Inspection
			Yard	None Identified	None Required
Valve Bodies	Pressure Boundary	Cast Iron	Raw Water	Loss of Material	Service Water Piping Corrosion Program Galvanic Susceptibility Inspection Selective Leaching Inspection
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Valve Bodies	Pressure Boundary	Carbon Steel	Raw Water	Loss of Material	Service Water Piping Corrosion Program Galvanic Susceptibility Inspection
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Based on the evaluations provided in Appendix B for the aging management programs and activities listed above, with the addition of the new activities described below and the additions to the **Scope** attributes for the select programs described below, the aging effects will be adequately managed such that the intended functions of the components listed in the above table will be maintained consistent with the current licensing basis for the period of extended operation.

The following information will be added to the **Scope** attribute of the summary descriptions for the programs in the Catawba UFSAR Supplement as follows:

Service Water Piping Corrosion Program- add Filtered Water System *Galvanic Susceptibility Inspection*- add Filtered Water System *Selective Leaching Inspection*- add Filtered Water System

The *Fire Protection Program – Jockey Pump Strainer Inspection* and *Fire Protection Program – Tank and Connected Piping Internal Inspection* are newly identified activities. A ten-attribute activity description and UFSAR Supplement revisions are provided for each activity below.

Fire Protection Program - Jockey Pump Strainer Inspection

Note: The JOCKEY PUMP STRAINER INSPECTION is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station.
The *Jockey Pump Strainer Inspection* is a new aging management activity. The purpose of the *Jockey Fire Pump Strainer Inspection* is to identify any loss of material of each stainless steel jockey pump strainer basket. A strainer is located at the suction side of each jockey pump. The raw water flow could result in loss of material of the strainer. This activity visually inspects the condition of the strainer baskets every ten years to check for loss of material. The *Jockey Pump Strainer Inspection* is a condition monitoring activity and is a new plant activity for license renewal.

Scope – The scope of the *Jockey Pump Strainer Inspection* is the strainer located on the suction side of each jockey pump.

Preventive Actions – No actions are taken as part of this program to prevent aging effects or to mitigate aging degradation.

Parameters Monitored or Inspected – The parameters inspected by the *Strainer Inspection* is loss of material due to exposure to a raw water environment.

Detection of Aging Effects – In accordance with information provided in **Monitoring & Trending** below, the *Jockey Pump Strainer Inspection* will detect loss of material of the jockey pump strainers prior to loss of component intended function.

Monitoring & Trending – The *Jockey Pump Strainer Inspection* is a general visual inspection for loss of material of the strainer baskets.

For McGuire, the initial *Jockey Pump Strainer Inspection* will be completed following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1).

For Catawba, the initial *Jockey Pump Strainer Inspection* will be completed following issuance of renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba Unit 1).

Acceptance Criteria – The acceptance criteria for the *Jockey Pump Strainer Inspection* is no unacceptable loss of material that could result in a loss of component intended function(s) as determined by engineering.

Corrective Action & Confirmation Process – If engineering evaluation determines that the observed aging effects do not cause a loss of component intended function, then no further actions are necessary. If engineering evaluation determines that the observed aging effects could cause a loss of component intended function, then corrective actions are taken, including cleaning of the strainer or replacement. Specific corrective actions will be implemented in accordance with the corrective action program.

Administrative Controls – The *Jockey Pump Strainer Inspection* will be implemented in accordance with controlled plant procedures.

Operating Experience – The *Jockey Pump Strainer Inspection* is a new inspection. Visual inspection is an effective method for detecting age-related degradation in the strainers. The strainers have been cleaned periodically through the years and loss of material has not been observed.

Proposed Revision to the UFSAR Supplements

The following statements will be added to the summary description of the *Fire Protection Program* in Section 18.2.8 of the McGuire UFSAR Supplement and Section 18.2.9 of the Catawba UFSAR Supplement:

> The *Jockey Pump Strainer Inspection* will identify any loss of material of each jockey pump strainer basket. The raw water flow could result in loss of material. The acceptance criteria for the *Jockey Pump Strainer Inspection* is no unacceptable loss of material that could result in a loss of component intended function(s) as determined by engineering.
>
> For McGuire, the initial *Jockey Pump Strainer Inspection* will be completed following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1).
>
> For Catawba, the initial *Jockey Pump Strainer Inspection* will be completed following issuance of renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba Unit 1).

Fire Protection Program - Tank and Connected Piping Internal Inspection

Note: The TANK AND CONNECTED PIPING INTERNAL INSPECTION is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station.

The purpose of the *Tank and Connected Piping Internal Inspection* is to manage loss of material of the internal surfaces of the carbon steel tanks and some connecting piping and valves in the Fire Protection System at McGuire and Catawba and the Filtered Water System at Catawba. The internal carbon steel surfaces of the tanks within the scope of this inspection are coated with an epoxy coating. Continued presence of an intact coating precludes loss of material of the internal surfaces of the carbon steel tanks that could lead to loss of pressure boundary function. This activity inspects the internal coating of the tanks every ten years to check the condition of the coating to identify coating failures and inspects some of the connected piping for loss of material. The *Tank and Connected Piping Internal Inspection* is a condition monitoring activity.

Scope – The scope of the *Tank and Connected Piping Internal Inspection* is the internal surface of the McGuire fire protection system pressure maintenance accumulator tank and the connecting piping and valves that supply high-pressure air. The scope of the program at Catawba is the equivalent fire protection system pressure maintenance accumulator tank. Additionally, at Catawba, the filtered water tanks and their connected aluminum piping in the supply system to the fire protection system will be inspected.

Preventive Actions – No actions are taken as part of this program to prevent aging effects or to mitigate aging degradation.

Parameters Monitored or Inspected – The *Tank and Connected Piping Internal Inspection* inspects the coating for signs of blistering, chipping, peeling, and missing coating as well as signs of corrosion of the underlying carbon steel tanks. The inspection also visually inspects the high-pressure air supply piping connected to the fire protection system pressure maintenance accumulator tank at McGuire and the aluminum piping connected to the filtered water tanks at Catawba for signs of loss of material. Due to the material and environment of this connected piping, little to no aging effects are expected in these latter components, which will be verified by this inspection.

Detection of Aging Effects – In accordance with the information provided under **Monitoring & Trending** below, the *Tank and Connected Piping Internal Inspection* will detect the condition of the tank coatings and any loss of material of connecting piping.

Monitoring & Trending – The *Tank and Connected Piping Internal Inspection* visually inspects the internal coating of the tanks. The inspection looks for signs of blistering, chipping, peeling, and missing paint as well as signs of corrosion of the underlying carbon steel tank. The inspection also visually inspects connecting piping described in **Parameters Monitored or Inspected** for signs of loss of material.

No actions are taken as part of this activity to trend inspection results.

For McGuire, the initial *Tank and Connected Piping Internal Inspection* will be completed following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1).

For Catawba, the initial *Tank and Connected Piping Internal Inspection* will be completed following issuance of renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba Unit 1).

Acceptance Criteria – The acceptance criteria for the *Tank and Connected Piping Internal Inspection* are no visual indications of coating defects that have led to corrosion of the underlying carbon steel tank surfaces and no unacceptable loss of material of the connecting piping that could result in an unacceptable loss of pressure boundary as determined by engineering evaluation. Unacceptable loss is defined using a high tolerance in this case since leakage of the pressure maintenance subsystem of the fire protection system can be tolerated and only serves to make the system less efficient and does not cause a failure in the ability of the system to accomplish the system function.

Corrective Action & Confirmation Process – Engineering evaluation is performed to determine whether the coating and base metal of the tank and the condition of the piping continue to be acceptable. If engineering evaluation determines that the observed aging effects could cause a loss of component intended function, then corrective actions are taken, including coating repairs or base metal or piping repair or replacement. Specific corrective actions and confirmation are implemented in accordance with the corrective action program.

Administrative Controls – The *Tank and Connected Piping Internal Inspection* is controlled by plant procedures and work processes. The procedures and work processes provide steps for performance of the activities and require documentation of the results.

Operating Experience – The *Tank and Connected Piping Internal Inspection* is a new inspection. Previous visual inspections of the McGuire tank and a similar tank at Catawba have demonstrated that visual inspection of internal surfaces is an effective method for detecting age-related degradation in the tanks and associated piping and valves.

Proposed Revision to the UFSAR Supplements

The following statements will be added to the summary description of the *Fire Protection Program* in Section 18.2.8 of the McGuire UFSAR Supplement:

> The purpose of the *Tank and Connected Piping Internal Inspection* is to manage loss of material of the internal surfaces of the carbon steel fire protection system pressure maintenance accumulator tank and connecting piping and valves supplying high-pressure air. The internal carbon steel surfaces of the tank are coated with an epoxy coating. Continued presence of an intact coating precludes loss of material of the internal surfaces of the carbon steel tank. This preventive maintenance activity inspects the internal coating of the fire protection system pressure maintenance accumulator tank to check the condition of the coating to identify coating failures and the condition of the connecting piping supplying high-pressure air to identify loss of material. The *Tank and Connected Piping Internal Inspection* is a condition monitoring activity.
>
> The initial *Tank and Connected Piping Internal Inspection* will be completed following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1).

The following statements will be added to the summary description of the *Fire Protection Program* in Section 18.2.9 of the Catawba UFSAR Supplement:

> The purpose of the *Tank and Connected Piping Internal Inspection* is to manage loss of material of the internal surfaces of the carbon steel fire protection system pressure maintenance accumulator tank and the filtered water tanks and connecting aluminum piping and valves. The internal carbon steel surfaces of the tanks are coated with an epoxy coating. Continued presence of an intact coating precludes loss of material of the internal surfaces of the carbon steel tanks. This preventive maintenance activity inspects the internal coating of the fire protection system pressure maintenance accumulator tanks and filtered water tanks to check the condition of the coating to identify coating failures and the condition of the connecting aluminum piping to identify loss of material. The *Tank and Connected Piping Internal Inspection* is a condition monitoring activity.
>
> The initial *Tank and Connected Piping Internal Inspection* will be completed following issuance of renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba Unit 1).

<u>Open Item 2.3.3.19-3</u> Duke did not identify Catawba fire suppression equipment to lower containment carbon filters as within the scope of license renewal. Section 9.5.1.2.1 of the UFSAR states that the RF system provides a fixed water suppression system for <u>charcoal</u> filters. On pages 48-50 of.Duke's revised response to Appendix A to BTP APCSB 9.5-1, submitted to the NRC by letter dated November 4, 1983, Duke stated that lower containment <u>carbon</u> filters are provided with fire suppression capability. According to NRC Inspection Report 50-369/02-05, 50-370/02-05, 50-413/02-05 and 50-414/02-05 (ML021280003), Duke Specification CNS-1465.00-00-0006 states that <u>carbon</u> filters are protected by built-in water spray systems. The staff does not believe that the applicant's distinction between charcoal and carbon filters is material. Therefore, the applicant should identify water suppression equipment associated with the protection of carbon (or charcoal) filters as within the scope of license renewal.

Duke Response to Open Item 2.3.3.19-3
Duke researched all of the documentation that comprises the current licensing basis to determine those systems, structures, and components required to comply with 10 CFR 50.48. Based on the staff's regulations and guidance, the success criteria of §50.48 is clearly the ability to safely shut down the plant and minimize radiation releases in the event of a design basis fire.

The UFSAR discusses the suppression of these filters. The fire protection SER discusses the existence of the suppression of these filters. The suppression for these filters is currently relied on in a plant evaluation to function so that radioactive releases are minimized. Because the suppression system is currently relied upon in a plant evaluation to perform a function to meet the success criteria of §50.48, it is within the scope of license renewal.

Further review by Duke has determined that the piping, sprinklers, and valve bodies associated with the Catawba Reactor Building Charcoal Filter Unit sprinklers should have been identified as within the scope of license renewal and subject to aging management review. The components of this portion of the Catawba Fire Protection System are listed in Table 3.3-27 of the Application. Please see Section 3.3.1 of the Application for a description of each column in this table. For the convenience of the reviewer, the aging management review results for this portion of the Catawba Fire Protection System are repeated below:

1	2	3	4	5	6
Component Type	**Component Function**	**Material**	**Internal Environment** **External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Pipe	Pressure Boundary	Carbon Steel	Ventilation	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Engineering Structures and Components
Sprinklers	Pressure Boundary & Spray	Brass	Ventilation	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program (Note 3)
Valve Bodies	Pressure Boundary	Stainless Steel	Ventilation	None Identified	None Required
			Reactor Building	None Identified	None Required

Note 3 is from LRA Table 3.3-27 and reads as follows:
The Fluid Leak Management Program is applicable only within the Reactor Building or Auxiliary Building.

Based on the evaluations provided in Appendix B for the aging management programs and activities listed above, the aging effects will be adequately managed such that the intended functions of the components listed in the above table will be maintained consistent with the current licensing basis for the period of extended operation.

Open Item 2.3.3.19-4 A license condition for McGuire and Catawba states that Duke Energy Corporation shall implement and maintain in effect all provisions of the approved fire protection program as described in the UFSARs for the respective facilities. Sections 9.5.1.2.1 and 9.5.1.2.2 of the UFSARs state that manual hose stations and automatic sprinkler or deluge systems are provided for the protection of oil storage house; the oxygen and acetylene gas storage yard area; compressed flammable gas cylinder storage area; main turbine piping and bearings; unit start-up and standby oil-filled power transformers; main turbine lube oil reservoirs; hydrogen seal oil unit; and the feedwater pump turbines. The UFSARs do not differentiate between those manual hose station and automatic sprinklers that are required to comply with 10 CFR 50.48 and those that are not. Additionally, the regulations governing fire protection apply to more than the protection of structures and equipment relied upon for safe plant shutdown. Therefore, the applicant should furnish documentation that demonstrates that the fire protection features are not required by 10 CFR 50.48 or identify the components associated with these manual hose stations and automatic sprinkler or deluge systems as being within the scope of license renewal and subject to an AMR.

Duke Response to Open Item 2.3.3.19-4
The staff is correct in identifying that the UFSAR does not differentiate between those sprinklers that are required for compliance with 10 CFR 50.48 and those that are not. Therefore, review of the UFSAR alone cannot determine those sprinklers required to comply with 10 CFR 50.48.

In a meeting between the staff and Duke on October 1, 2002, the staff indicated that sufficient information was not readily available to determine the licensing basis of the suppression systems in the areas listed in the open item. The staff's concern with the suppression systems in these outlying areas is that an exposure hazard may exist that may exceed the 3-hour capability of the fire barrier surrounding the Auxiliary Building and that during original licensing, the staff may have relied on suppression in these outlying areas (particularly manual suppression in the Turbine Building) to provide programmatic defense in depth to ensure the capability to achieve safe shutdown and to minimize radioactive releases in the event of a fire in these areas.

Duke agreed to research the licensing basis and report the findings in the response to Open Item 2.3.3.19-4. In addition to researching the licensing basis, Duke reviewed each area listed in the open item to validate whether an exposure hazard does exist that would jeopardize areas of the Auxiliary Building that house safety-related and safe shutdown equipment. The results of the review are provided below for Catawba and McGuire separately.

Catawba

Catawba's Fire Protection Review (FPR) consists of submittals made to the NRC during original licensing that includes the response to Appendix A to Branch Technical Position APCSB 9.5-1 and the Fire Hazards Analysis (FHA). These submittals were made several times during the licensing process. The submittals of record when the SER was issued are made in the following letters:

- Letter from W. O. Parker, Jr. (Duke) to Harold R. Denton (NRC), *Catawba Nuclear Station Fire Protection Review*, October 23, 1981.
- Letter from Hal B. Tucker (Duke) to Harold R. Denton (NRC), *Catawba Nuclear Station Fire Protection Review*, November 4, 1983.

The NRC staff's SER compared Catawba to NUREG-0800 which contains Branch Technical Position CMEB 9.5-1.

Duke has reviewed the entire Fire Protection Review and the subsequent SER to achieve a holistic understanding of the information the staff may have relied on to make a finding that the fire protection program is acceptable. The guidelines of Appendix A to Branch Technical Position APCSB 9.5-1 contain the elements of defense in depth of the fire protection program. The table below provides the information from Appendix A to Branch Technical Position APCSB 9.5-1, Duke's Fire Protection Review, the Catawba SER and comments that provide the perspective gained by reviewing all the applicable licensing documents.

Review of the licensing historical information reveals that the staff did not explicitly rely on suppression in these outlying areas as a defense-in-depth aspect of the fire protection program. The licensing basis credits the 3-hour fire barrier to protect equipment in the Auxiliary Building relied on to ensure safe plant shutdown and minimize radioactive releases from a fire in these outlying areas. The Fire Protection Review information is kept current by inclusion in the Catawba Design Basis Document for Fire Protection. The information in the current Catawba Fire Protection Design Basis Document is the same as the original response to the BTP and the Fire Hazards Analysis, indicating there have been no program changes with respect to this issue.

In addition to performing the licensing basis review, Duke performed a re-review of the designs that formed the basis of the original licensing basis. For those areas listed in the open item that are located in the yard, each area is hundreds of feet from the Nuclear Service Water Pump Structure and the fire barrier that separates the Auxiliary Building and Reactor Buildings from the yard. A fire in these areas of the yard does not present an exposure hazard to the Nuclear Service Water Pump Structure, Auxiliary Building, or Reactor Buildings. For those areas that are located in the Turbine Building, the limiting area is associated with the main turbine components. The Main Turbine Lubricating Oil Tank, which contains the largest volume of combustible fluid in the Turbine Building, is located approximately 100 feet from the fire barrier that separates the Auxiliary Building from the Service Building and Turbine Building. Thus, these areas do not present an exposure hazard to the Auxiliary Building.

In conclusion, based on a review of the Fire Protection Program licensing information and technical review of plant design, Duke concludes that suppression in outlying plant areas and the Turbine Building are not within the scope of license renewal

Attachment 2

Response to
McGuire Units 1 & 2 and Catawba Units 1 & 2
Safety Evaluation Report with Open Items

Mechanical Related Items

Catawba Licensing Review Information					
Issue	**Appendix A to BTP APCSB 9.5-1 Reference**	**FPR Reference (FHA)**	**FPR Reference (Response to BTP)**	**SER Reference**	**Duke Comments**
Turbine Bldg, Service Bldg, Admin Bldg, Oil Storage House, Gas Cylinder Storage Areas, Transformer Yard, Turbine Lube Oil areas, etc	**C.7.h** (Turbine Building) "The turbine building should be separated from adjacent structures containing safety related equipment by a fire barrier with a minimum rating of 3 hours......Considering the severity of the fire hazards, defense in depth **may** dictate additional protection to ensure barrier integrity." **C.7.r** (Miscellaneous Areas) "Miscellaneous areas such as shops, warehouses, auxiliary boiler rooms, fuel oil tanks, and flammable and combustible liquid storage tanks should be so located and protected that a fire and effects of a fire....will not adversely affect any safety-related systems or equipment."	**FHA 10/23/81 pg C-2** **FHA 11/04/83 pg C-2** "The analysis was conducted in the Auxiliary, Diesel, Reactor Building, and Nuclear Service Water Pump Structure and that portion of the Service Building which is adjacent to the Auxiliary Building "	**FPR 10/23/81 pgs 85, 91** **FPR 11/04/83 pgs 91, 97** "The Turbine and Auxiliary Buildings are separated by a three-hour barrier; therefore, the turbine oil system is separated from all safety related equipment." "The fire hazard analysis was a primary medium for determining that safe shutdown equipment was isolated from unacceptable fire hazards, including those listed as Miscellaneous Areas."	**SER pg 9-45, 9-46** "The applicant's Fire Hazards Analysis addressed other station areas not specifically discussed in this report. The staff finds that the fire protection for these areas is in accordance with the guidelines of BTP 9 5-1, Item **C.7**, and is, therefore, acceptable "	The key element of Catawba's CLB with respect to Fire Protection features in areas addressed by Open Item 2.3.3.19-4 is given in the Response to Appendix A to BTP APCSB 9 5-1 given in the FPR The FHA and our CLB only credits the separation provided by the 3-hour barriers The FHA introductory statement explains the areas of the plant that the fire hazard analysis was performed for. This includes "exposure hazards" from the Service Building (Note that the Turbine Building is not directly adjacent to the Auxiliary Building at Catawba). The FHA and the response to the BTP did NOT indicate a need or reliance on "defense in depth" to "ensure barrier integrity." The current Fire Protection Plant DBD has the same information with respect to the Response to the BTP and the FHA – indicating there have been no program changes with respect to this issue
Concern regarding Turbine Building manual hose stations raised during meeting with staff on 10/1/02	**C.6.c(4)** "Interior manual hose installation should be able to reach any location that contains, or could prevent a fire exposure hazard to, safety-related equipment with at least one effective hose stream ... Hose stations should be located as dictated by the fire hazard analysis....."	**FHA 10/23/81 pg C-2** **FHA 11/04/83 pg C-2** "The analysis was conducted in the Auxiliary, Diesel, Reactor Building; and Nuclear Service Water Pump Structure and that portion of the Service Building which is adjacent to the Auxiliary Building."	**FPR 10/23/81 pg 70** **FPR 11/04/83 pg 76** "Interior manual hose installations are provided and equipped to reach any location with at least one effective hose stream "	**SER pg 9-42** "Interior manual hose stations are provided and equipped to reach any plant location with at least one effective hose stream.. ...The staff finds that the hose stations meet the guidelines of BTP CMEB 9.5-1, Item C 6 c, and are, therefore, acceptable."	The referenced portion of the FHA is the only Catawba CLB document that discusses fire exposure hazards with respect to potential impact to safety-related structures and components and safe shutdown capability and mitigation of radioactive release. The FHA only makes mention of the Service Building and does not recognize the Turbine Building as a fire exposure hazard. The FHA does not evaluate the Service Building as an exposure hazard and therefore, only credits the 3 hour barrier between the Service Building and Auxiliary Building. Manual hose streams outside of safety-related areas are not credited to mitigate a potential fire exposure hazard. The current Fire Protection Plant DBD has the same information with respect to the Response to the BTP and the FHA – indicating there have been no program changes with respect to this issue.

McGuire

McGuire's Fire Protection Review consists of submittals made to the NRC that includes the response to Appendix A to Branch Technical Position APCSB 9.5-1 and the Fire Hazards Analysis. These submittals were made several times during the licensing process. The submittal of record when the SER was issued is made in the following letter:

> Letter from W. O. Parker, Jr. (Duke) to Harold R. Denton (NRC), *McGuire Nuclear Station Fire Protection Review*, January 31, 1979.

Duke has reviewed the entire Fire Protection Review and the subsequent SER to achieve a holistic understanding of the information the staff may have relied on to make a finding that the fire protection program is acceptable. The guidelines of Appendix A to Branch Technical Position APCSB 9.5-1 contain the elements of defense in depth of the fire protection program. The table below provides the information from Appendix A to Branch Technical Position APCSB 9.5-1, Duke's Fire Protection Review, the McGuire SER and comments that provide the perspective gained by reviewing all the applicable licensing documents.

Review of the licensing information reveals that the staff did not explicitly rely on suppression in these outlying areas as a defense-in-depth aspect of the fire protection program. The licensing basis credits the 3-hour fire barrier to protect equipment relied on to ensure safe plant shutdown and minimize radioactive releases from a fire in these outlying areas. The Fire Protection Review information is kept current by inclusion in the McGuire Design Basis Document for Fire Protection. The information in the current McGuire Fire Protection Design Basis Document is the same information as the original response to the BTP and the Fire Hazards Analysis, indicating there have been no program changes with respect to this issue.

In addition to performing the licensing basis review, Duke performed a re-review of the designs that formed the basis of the original licensing basis. For those areas listed in the open item that are located in the yard, each area is hundreds of feet from the fire barrier that separates the Auxiliary Building and Reactor Buildings from the yard. A fire in these areas of the yard does not present an exposure hazard to either the Auxiliary Building or Reactor Buildings. For those areas that are located in the Turbine Building, the limiting area is associated with the main turbine components. The Turbine Lubricating Oil Storage Tank is located approximately 20 feet from the Auxiliary Building wall. This wall is constructed of 3-feet thick reinforced concrete, which is conservatively designated as a 3-hour fire barrier. There are no penetrations in the wall in close proximity to the Turbine Lubricating Oil Storage Tank. Thus, these areas do not present an exposure hazard to the Auxiliary Building.

In conclusion, based on a review of the Fire Protection Program licensing information and technical review of plant design, Duke concludes that suppression in outlying plant areas and the Turbine Building are not within the scope of license renewal.

McGuire Licensing Review Information					
Issue	**Appendix A to BTP APCSB 9.5-1 Reference**	**FPR Reference (FHA)**	**FPR Reference (Response to BTP)**	**SER Reference**	**Duke Comments**
Turbine Bldg, Service Bldg, Admin Bldg, Oil Storage House, Gas Cylinder Storage Areas, Transformer Yard, Turbine Lube Oil areas, etc	**F.8** (Turbine Lubrication and Control Oil Storage and Use Areas) "A blank fire wall having a minimum resistance rating of three hours should separate all areas containing safety related systems and equipment from the turbine oil system." **F.18** (Miscellaneous Areas) "Miscellaneous areas such as records storage areas, shops, warehouses, and auxiliary boiler rooms should be so located that a fire or effects of a fire, including smoke, will not adversely affect any safety-related systems or equipment. Fuel oil tanks for auxiliary boilers should be buried or provided with dikes to contain the entire tank contents."	**FHA 1/31/79 pg C-2** "The analysis was conducted in the Auxiliary and Reactor Buildings and that portion of the Turbine and Service Buildings which are adjacent to the Auxiliary Building."	**FPR 1/31/79 pg 75** "The Turbine and Auxiliary Buildings are separated by a three-hour barrier; therefore, the turbine oil system is separated from all safety related equipment." **FPR 1/31/79 pg 81** "The fire hazard analysis was a primary medium for determining that safe shutdown equipment was isolated from unacceptable fire hazards, including those listed as Miscellaneous Areas "	**SER Supplement 5 pg B-10** "The applicant's Fire Hazards Analysis addresses other plant areas not specifically discussed in this report. The applicant has committed to install additional detectors, portable extinguishers, hose stations, and some additional emergency lighting as identified in the applicant's installation schedule. We find these areas with the commitment made by the applicant to be in accordance with the guidelines of Appendix A of BTP 9.5-1, and the applicant sections of the National Fire Protection Association Code and are therefore acceptable."	The key element of McGuire's CLB with respect to Fire Protection features in areas addressed by Open Item 2 3 3.19-4 is given in the Response to Appendix A to BTP APCSB 9 5-1 given in the FPR. The FHA and our CLB only credits the separation provided by the 3-hour barriers The FHA introductory statement explains the areas of the plant that the fire hazard analysis was performed for. This includes "exposure hazards" from the Turbine Building and Service Building. The FHA and the response to the BTP did NOT indicate a need or reliance on "defense in depth" to "ensure barrier integrity." The current Fire Protection Plant DBD has the same information with respect to the Response to the BTP and the FHA – indicating there have been no program changes with respect to this issue The installation schedule referred to in the SER did not include any suppression in the Turbine Building or outlying areas.

McGuire Licensing Review Information					
Issue	**Appendix A to BTP APCSB 9.5-1 Reference**	**FPR Reference (FHA)**	**FPR Reference (Response to BTP)**	**SER Reference**	**Duke Comments**
Concern regarding Turbine Building manual hose stations raised during meeting with staff on 10/1/02	**E.3.d** "Interior manual hose installation should be able to reach any location with at least one effective hose stream....."	**FHA 1/31/79 pg C-2** "The analysis was conducted in the Auxiliary and Reactor Buildings and that portion of the Turbine and Service Buildings which are adjacent to the Auxiliary Building."	**FPR 1/31/79 pg 63** "Interior manual hose installations are provided and equipped to reach any location with at least one effective hose stream."	**SER Supplement 5 pg B-3** "Manual hose stations are located throughout the plant to ensure that an effective hose stream can be directed to any safety related area in the plant"	The referenced portion of the FHA is the only McGuire CLB document that discusses fire exposure hazards with respect to potential impact to safety-related structures and components and safe shutdown capability and mitigation of radioactive release. The FHA does not evaluate the Turbine Building and Service Building as an exposure hazard and therefore, only credits the 3 hour barrier between the turbine and Service Buildings and the Auxiliary Building. Although the individual section in the BTP does not delineate this requirement for protecting safety-related equipment, it is obviously the intent as evidenced by the SER statement and the wording of the section in the later SRP version of the BTP. Manual hose streams outside of safety-related areas are not credited to mitigate a potential fire exposure hazard The current Fire Protection Plant DBD has the same information with respect to the Response to the BTP and the FHA – indicating there have been no program changes with respect to this issue

Open Item 2.3.3.19-5 The staff agrees with the applicant that the strainers perform an intended function that meets one of the scoping criteria (specifically 10 CFR 54.4(a)(3). The staff's technical concern is that Duke uses lake water to supply their fire protection suppression systems at McGuire and Catawba. Lake water is corrosive and may contain sediment, which can potentially clog the fire pumps. In addition, the strainers keep debris from plugging the sprinkler nozzles in fire suppression systems in the event that sprinklers are actuated. This fire protection component should be managed in an AMP. However, the staff is concerned that the strainers were inappropriately screened out. Although the strainers may be in-line with and connected to the main fire pump, their function is passive (as is the pump casing's function). The applicant included the pump casings within the scope of license renewal; the strainers also should be within scope.

Duke Response to Open Item 2.3.3.19-5
Duke will include the main fire pump strainers as subject to aging management review. Provided below is the aging management review for the main fire pump strainers.

Note: The aging management review of the strainer is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station.

Each station has three main fire pumps. The pumps are normally in standby and are automatically started on low system pressure. Each pump has a strainer that is within the scope of license renewal and is subject to aging management review because it is long-lived, passive component. The strainer prevents debris from entering the pump when it is in operation thus protecting the pump from damage. The strainer has a ½ inch mesh and can be made of either bronze or stainless steel. In order to manage the effects of aging, a new inspection will be implemented. The following is a summary of the aging management review:

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material**	**Internal Environment** / **External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Main Fire Pump Strainers	Filtration	Bronze or Stainless Steel	Raw Water Note (2)	Loss of Material	Fire Protection Program – Main Fire Pump Strainer Inspection

Notes:

(1) Filtration – Provide filtration of process fluid so that downstream equipment and/or environments are protected.
(2) The Main Fire Pump Strainers are located on the suction side of the pumps, totally immersed in raw water.

Main Fire Pump Strainer Inspection

The *Main Fire Pump Strainer Inspection* is a new aging management activity. The purpose of the *Main Fire Pump Strainer Inspection* is to identify any loss of material of each main fire pump strainer. The strainer is attached to the base of the suction bell of each pump. The raw water flow could result in loss of material of the strainer. The *Main Fire Pump Strainer Inspection* will inspect the strainers for loss of material at least once every ten years.

Scope – The scope of the *Main Fire Pump Strainer Inspection* is the strainer located on the suction bell of each main fire pump.

Preventive Actions – No actions are taken as part of this program to prevent aging effects or to mitigate aging degradation

Parameters Monitored or Inspected – The parameters inspected by the *Strainer Inspection* is loss of material due to exposure to a raw water environment.

Detection of Aging Effects – In accordance with information provided in **Monitoring & Trending** below, the *Main Fire Pump Strainer Inspection* will detect loss of material of the main fire pump strainers prior to loss of component intended function.

Monitoring & Trending – The *Main Fire Pump Strainer Inspection* is a general visual inspection for loss of material of the strainer. The *Main Fire Pump Strainer Inspection* will be performed at least once every ten years.

For McGuire, the initial *Main Fire Pump Strainer Inspection* will be completed following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1).

For Catawba, the initial *Main Fire Pump Strainer Inspection* will be completed following issuance of renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba Unit 1).

Acceptance Criteria – The acceptance criteria for the *Main Fire Pump Strainer Inspection* is no unacceptable loss of material that could result in a loss of component intended function(s) as determined by engineering.

Corrective Action & Confirmation Process – If engineering evaluation determines that the observed aging effects do not cause a loss of component intended function, then no further actions are necessary. If engineering evaluation determines that the observed aging effects could cause a loss of component intended function, then corrective actions are taken, including cleaning of the strainer or replacement. Specific corrective actions will be implemented in accordance with the corrective action program.

Administrative Controls – The *Main Fire Pump Strainer Inspection* will be implemented in accordance with controlled plant procedures.

Operating Experience – The *Main Fire Pump Strainer Inspection* is a new inspection for which there is no operating experience. The inspection frequency is based on the planned frequency for performing routine maintenance on each main fire pump.

Proposed Revision to the UFSAR Supplements

The following statements will be added to the summary description of the Fire Protection Program in each station's UFSAR Supplement:

> The *Main Fire Pump Strainer Inspection* will identify any loss of material of each main fire pump strainer. The raw water flow could result in loss of material. The acceptance criteria for the *Main Fire Pump Strainer Inspection* is no unacceptable loss of material that could result in a loss of component intended function(s) as determined by engineering.
>
> For McGuire, the initial *Main Fire Pump Strainer Inspection* will be completed following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1).
>
> For Catawba, the initial *Main Fire Pump Strainer Inspection* will be completed following issuance of renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba Unit 1).

New Open Item 2.3.3.19-6 10 CFR 50.48 requires each operating nuclear station to have a fire protection plan. A license condition for McGuire and Catawba states that Duke Energy Corporation shall implement and maintain in effect all provisions of the approved fire protection program as described in the UFSAR for the respective facilities. Section 9.5.1.2.3, "Fire Protection, Category I Safety Related," of the McGuire UFSAR states that the manually operated water spray systems provide fixed spray patterns of water for Reactor Building Purge Exhaust Filters 1A, 1B, 2A and 2B. However, drawing MCFD 1599-02.01, coordinates H-3, G-3, C-5 and B-7, indicates that piping and sprinklers associated with this function are also excluded from scope. The staff is concerned that the manually operated water spray systems for these filters were inappropriately excluded from the scope of license renewal and an AMR.

Duke Response to Open Item 2.3.3.19-6
Duke researched all of the documentation that comprises the current licensing basis to determine those systems, structures, and components required to comply with 10 CFR 50.48. Based on the staff's regulations and guidance, the success criteria of §50.48 is clearly the ability to safely shut down the plant and minimize radiation releases in the event of a design basis fire.

The UFSAR discusses the suppression of these filters. The fire protection SER discusses the existence of the suppression of these filters. The suppression for these filters is currently relied on in a plant evaluation to function so that radioactive releases are minimized. Because the suppression system is currently relied upon in a plant evaluation to perform a function to meet the success criteria of §50.48, it is within the scope of license renewal.

Further review by Duke has determined that the flexible hoses, piping, sprinklers, and valve bodies associated with the McGuire Reactor Building Exhaust filters spray system should have been identified as within the scope of license renewal and subject to aging management review. The components of this portion of the McGuire Fire Protection System are listed in Table 3.3-26 of the Application. For the convenience of the reviewer, the aging management review results for this portion of the McGuire Fire Protection System are repeated below:

1	2	3	4	5	6
Component Type	**Component Function**	**Material**	**Internal Environment** / **External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Flexible Hose	Pressure Boundary	Stainless Steel	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Pipe	Pressure Boundary	Galvanized Steel or Carbon Steel	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Engineering Structures and Components
Rupture Disk	Pressure Boundary	Carbon Steel	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Engineering Structures and Components
Spray Nozzles	Pressure Boundary & Spray	Bronze	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3)
Spray Nozzles	Pressure Boundary & Spray	Carbon Steel	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Engineering Structures and Components
Spray Nozzles	Pressure Boundary & Spray	Stainless Steel	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required

1	2	3	4	5	6
Component Type	**Component Function**	**Material**	**Internal Environment** / **External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Valve Bodies	Pressure Boundary	Cast Iron or Carbon Steel	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Engineering Structures and Components

Note 3 is from LRA Table 3.3-26 and reads as follows:
The Fluid Leak Management Program is applicable only within the Reactor Building or Auxiliary Building.

Based on the evaluations provided in Appendix B for the aging management programs and activities listed above, the aging effects will be adequately managed such that the intended functions of the components listed in the above table will be maintained consistent with the current licensing basis for the period of extended operation.

Open Item 2.3.3.35.2-1 The applicant did not provide sufficient information in its response to RAI 2.3.3.35-3 to enable the staff to evaluate the adequacy of its replacement of flexible hose connections associated with the standby shutdown diesel generator fuel oil sub-system during periodic maintenance. The applicant should indicate if replacement of these components is based upon a qualified life or based upon condition or performance monitoring. If replacement is based upon the latter, the applicant should specify the parameters that will be monitored as indicators of the components' condition or performance.

Duke Response to Open Item 2.3.3.35.2-1
The flexible hoses on the Standby Shutdown Diesel Generator Fuel Oil Sub-system are replaced based upon condition. Every eighteen months, the flexible hoses are inspected for cracking and signs of wear.

New Open Item 3.0.3.2.3-1 The applicant provided in Appendix A-1 (McGuire) and A-2 (Catawba) new FSAR sections describing the chemistry control program. The information provided for the FSAR is consistent with the program described in Appendix B; however, the applicant should include a discussion in the FSAR Supplement regarding the specific technical specifications and the EPRI guidelines that are mentioned in Appendix B for the chemistry control program.

Duke Response to New Open Item 3.0.3.2.3-1

In response to New Open Item 3.0.3.2.3-1, the summary description of the *Chemistry Control Program* in each station's UFSAR Supplement will be revised to include the following statement:

> The *Chemistry Control Program* contains system specific acceptance criteria that are based on the guidance provided in EPRI PWR Primary Water Chemistry Guidelines, EPRI PWR Secondary Water Chemistry Guidelines, and EPRI Closed Cooling Water Chemistry Guideline.

The *Chemistry Control Program* entry of Table 18-1 of the McGuire UFSAR Supplement will be revised to read as follows:

Topic	***Application Location***	***UFSAR / ITS Location***
Chemistry Control Program	B.3.6	18.2.4 ITS 5.5.10 ITS 5.5.13 SLC 16.5-7 SLC 16.8-3 SLC 16.9-7

The *Chemistry Control Program* entry of Table 18-1 of the Catawba UFSAR Supplement will be revised to read as follows:

Topic	***Application Location***	***UFSAR / ITS Location***
Chemistry Control Program	B.3.6	18.2.4 ITS 5.5.10 ITS 5.5.13 SLC 16.5-3 SLC 16.7-9 SLC 16.8-5

New Open Item 3.0.3.9.1.2(a-g) The applicant's acceptance criteria for heat exchanger preventive maintenance are not adequate to provide the staff with reasonable assurance that loss of material of the heat exchanger components will be adequately managed or monitored such that the intended functions of the heat exchangers will be maintained during the extended period of operation. This open item applies to seven aging management activities (a through f).

Duke Response to New Open Item 3.0.3.9.1.2(a-g)
New open item 3.0.3.9.1.2(a-g) applies to the following heat exchangers:

Open Item Number	Component	SER Section Number	LRA Section Number
3.0.3.9.1.2(a)	Pump Motor Air Handling Units (McGuire only)	3.0.3.9.1.2	B.3.17.6
3.0.3.9.1.2(b)	Pump Oil Coolers (McGuire only)	3.0.3.9.2.2	B.3.17.7
3.0.3.9.1.2(c)	Containment Spray Heat Exchangers	3.2.4.2.2	B.3.17.2.2
3.0.3.9.1.2(d)	Component Cooling Heat Exchangers	3.3.5.2.2	B.3.17.1.2
3.0.3.9.1.2(e)	Control Area Chilled Water Chillers	3.3.8.2.2	B.3.17.4
3.0.3.9.1.2(f)	Diesel Generator Engine Cooling Water Heat Exchangers	3.3.12.2.2	B.3.17.3.2
3.1.3.9.1.2(g)	Diesel Generator Engine Starting Air Aftercoolers (Catawba only)	3.3.17.2.2	B.3.17.5

The acceptance criteria for each Heat Exchanger Preventive Maintenance Program are no unacceptable loss of material that could result in a loss of the component intended function as determined by engineering evaluation. Duke agrees that additional information describing the engineering evaluation that will be used to define "unacceptable loss of material" is needed for the staff to make a reasonable assurance finding with respect to acceptance criteria of the programs. The following details of each Heat Exchanger Preventive Maintenance Program are provided to assist in the finding.

For New Open Item 3.1.3.9.1.2(a), the program credited for managing loss of material for the pump motor air handling units is a new program to be implemented following the issuance of the renewed operating license for McGuire and by June 12, 2021. Because these heat exchanger tubes are a coil design, they are not candidates for eddy current testing. As described in Section B.3.17.6 of the LRA, either destructive or nondestructive examination will be performed that allows examination of the internal surfaces of the tubes. If evidence of loss of material is observed during the initial inspection, a problem report will be developed in accordance with Problem Investigation Process of Nuclear System Directive 208. The Problem Investigation Process is a formalized process for documenting engineering evaluations of plant problems that would include the assessment of the severity of the observed degradation, the need for corrective actions, the need for further inspections of other locations, and the need for future inspections or programmatic oversight. Criteria such as ASME Code requirements, additional inspection results, and operating experience may be used to assess the severity of the degradation and the need for corrective actions. Any criteria or analysis methods involved in determining the severity of the degradation and the need for corrective action will be developed at the time of the evaluation and will be a part of the problem report. Duke believes it is premature to specify actual criteria for evaluating severity and the need for corrective actions for a new inspection for which the analysis method is not yet known.

For New Open Items 3.1.3.9.1.2(b) through 3.1.3.9.1.2(g), eddy current testing is the method used to manage loss of material of the heat exchanger tubes. The information that follows describes the acceptance criteria that apply to the existing programs which are the subject of 3.1.3.9.1.2(c) through (g) and will apply to the new program that is the subject of New Open Item 3.1.3.9.1.2(b).

Eddy current testing is an acceptable industry practice used for detecting wall loss in heat exchangers, but requires careful engineering evaluation of all test results to provide the proper management of a heat exchanger. Steam Generators are the only plant heat exchangers for which there exists station Technical Specifications or set of standards that regulate the depth of flaw at which a tube is plugged and removed from service. For the low pressure, low temperature heat exchangers that are the subject of these open items, evaluating eddy current test results for "unacceptable loss of material" involves many variables such as tube material, characterization of the indication in terms of percent wall loss, rate of degradation as compared to previous indications and the frequency of subsequent testing. A greater wall loss range may be considered acceptable for an indication that is tested frequently or that shows little or no degradation from previous tests; a lesser wall loss range may be considered unacceptable if the indication shows significant degradation from previous tests or that is not tested as frequently. Criteria such as ASME Code requirements, additional inspection results, and operating experience may be used to assess the severity of the degradation and the need for corrective actions.

Eddy current testing at Duke is performed by a vendor who specializes in the practice. The vendor supplies an eddy current test report to Duke for each test they perform. The four-step acceptance criteria process described below is used to generate the final test report.

(1) At the conclusion of testing of a component, the vendor's eddy current testing manager reviews the data and makes a plugging recommendation in the preliminary report based on his assessment of the damage flaws and experience with testing the component. Experience demonstrates that these specialists generally recommend evaluation at around a 70% wall loss range.

(2) Duke then reviews the entire test data provided in the preliminary test report, including the recommendation for plugging, prior to returning the component to service. Duke evaluates the recommendations using all the information they have available. Particularly, Duke evaluates the rate of degradation based on the history of the tube. The wall loss may be deemed acceptable if the tube is showing minimal to no degradation from previous inspections. Consideration is also given to the frequency of the next inspection; if frequent inspection is performed, then a higher wall loss range may be acceptable and if less frequent inspection is performed then lower wall loss range may be unacceptable.

(3) Depending on the type of tubing material and tubing damage detected with eddy current testing and possibly verified with actual tube pulled samples, a wall loss correlation may be determined as a threshold for evaluating the tube for plugging repair. Past operating experience with the type of tubing flaw may also be a very useful factor in determining the wall loss plugging threshold.

(4) The loss of material experienced by these heat exchanger tubes generally manifests itself as pits. These pitting flaws are not very likely to fail heat exchanger tubing due to mechanical stress of pressure and temperature due to the shouldered nature or material reinforcement around pits. Therefore, the pitting rate as determined from past eddy current testing experience becomes the primary factor to consider when selecting tubes to remove from service to prevent later on-line tube leaks.

Duke's experience in evaluating eddy current testing results has proven effective during the operation of McGuire and Catawba. Corrective actions such as tube plugging and even tube bundle and heat exchanger replacement have been taken as a result of failed acceptance criteria of these programs. Duke's experience of using the four-step process of evaluating "no unacceptable loss of material" described above provides reasonable assurance that the aging effects of these heat exchanger subcomponents will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

The summary descriptions each of the *Heat Exchanger Activities* contained in each station's UFSAR Supplement will be revised to include the following statement:

> Criteria such as ASME Code requirements, additional inspection results, and operating experience may be used to assess the severity of the degradation and the need for corrective actions.

Duke Identified Mechanical Item 09/18/2002

In the process of reviewing the SER, Duke identified a statement in Section B.3.17.7, *Heat Exchanger Preventive Maintenance Activities- Pump Oil Coolers* that does not clearly reflect Duke's intention with respect to these coolers. The *Heat Exchanger Preventive Maintenance Activities- Pump Oil Coolers* is a new program for license renewal and applies to eight (8) coolers per unit for a total of sixteen (16) coolers at McGuire only. The program description in the LRA states "Non-destructive (NDT) will be performed on 100% of the tubes," which could be misinterpreted. Duke's intention is to perform non-destructive testing on 100% of the tubes of one of the sixteen coolers within the scope of the program.

The selection of the specific cooler to be examined will take into consideration the normal operating environments of the coolers. The reciprocating charging pump and safety injection pump do not run during normal operation and therefore the reciprocating charging pump bearing oil coolers and speed reducing oil coolers and the safety injection pump bearing oil coolers are normally isolated. The centrifugal charging pumps are normally in service and therefore the centrifugal charging pump bearing oil coolers and speed reducing oil coolers have raw water flowing through them during normal operation. Duke's operating experience from performing periodic maintenance of all of the coolers has indicated that the coolers not normally in service are typically found to be in good condition with minimal fouling, while the coolers in operation experience more fouling buildup. It is expected that the development of corrosion products would be more likely in the fouled coolers than those with minimal fouling simply because of the more significant "under deposit" environment. Therefore, the centrifugal charging pump bearing oil coolers and speed reducing oil coolers should experience the most susceptible service environment for loss of material to occur. One of the centrifugal charging pump's coolers will therefore be examined as a representative of the total scope.

A sample inspection of one of the sixteen coolers is considered acceptable because of the excellent operating experience of these coolers. As described under Operating Experience in Section B.3.17.7 of the LRA, there have been no tube failures in any of the heat exchangers within the scope of this program, as confirmed through periodic leak detection. This leak detection is performed via periodic oil sampling. Additionally, no indications of tube pitting have been seen during periodic maintenance. The sample chosen is appropriate as a leading indicator of other components in the program because it is most likely to experience aging effects. Prior experience in leak detection provides a basis for concluding that the program will be an effective method of monitoring the components during the period of extended operation. Therefore, past operating experience can be relied on to provide the basis for this new program.

Implementing a sample inspection as part of the *Heat Exchanger Preventive Maintenance Activities- Pump Oil Coolers* more closely aligns this program with the *Heat Exchanger Preventive Maintenance Activities- Pump Motor Air Handling Units* described in Section B.3.17.6 of the LRA and found acceptable by the staff in Section 3.0.3.9.1.2 of the SER.

A review of the McGuire UFSAR Supplement revealed that a more clear description should be included for *Pump Oil Coolers* and *Pump Motor Air Handling Units.* The McGuire UFSAR Supplement summary description of the *Pump Oil Coolers* will be revised to add the following statement:

A non-destructive examination will be performed on 100% of the tubes of one of the sixteen coolers within the scope of the program following issuance of renewed licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1).

In addition, the McGuire UFSAR Supplement summary description of the *Pump Motor Air Handling Units* will be revised to add the following statement:

A destructive or non-destructive examination will be performed on one of the twelve cooling units within the scope of the program following issuance of renewed licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1).

New Open Item 3.0.3.13.2-1 In the case of the buried piping, the staff finds the applicant's Preventive Maintenance Activities - Condenser Circulating Water System Internal Coating Inspection program ineffective at revealing degradation of the external pipe surface before the component pressure boundary is breached and leakage occurs. The staff believes that the applicant should propose an activity to verify that the external surfaces of buried components are not degrading based upon some sampling assessment of most vulnerable locations.

Duke Response to New Open Item 3.0.3.13.2-1

In an electronic communication dated September 23, 2002, the NRC staff provided the following:

> The staff has re-evaluated Open Item 3.0.3.13.2-1 and considers it resolved for the following reasons:
>
> (1) Corrosion of the outside surface of a buried pipe occurs at locations where the coating is damaged and this can happen anywhere along the pipe. In order to obtain meaningful information, the whole length of the pipe would need to be excavated, which is not practical.
>
> (2) If a leak develops due to corrosion of the outside of pipe (because of damage to the outside coating), this leakage would affect the internal coating (e.g. blisters or other type of damage would be easily recognizable signs of damage to the internal coating). Inspection of the inside coating will reveal, therefore, the location of the leak.
>
> (3) Degree of degradation of the inside coating can give some idea on the condition of the outside coating.

Therefore, no further response from Duke is required.

New Open Item 3.0.3.15.2-1 In its description of the Service Water Piping Corrosion program, Monitoring and Trending element, the applicant stated that localized corrosion due to pitting and MIC will reveal itself through pinhole leaks in the piping components, that they are not a structural integrity concern, and that they cannot individually lead to loss of the component intended function, since sufficient flow at prescribed pressures can still be provided by the system. The applicant also state that these localized concerns will lead to structural integrity concerns only when a significant number of pinholes are present and that a trend of indications of through-wall leaks will trigger corrective actions. However, the staff believes that localized corrosion can result in the loss of pressure boundary intended function under a design basis event before the corrosion reveals itself as pinhole leaks. Therefore, the applicant should justify how its program will manage the effects of localized corrosion from pitting and MIC to ensure that the intended pressure boundary function can be maintained under all design basis events consistent with the CLB during the period of extended operation, as required by 10 CFR 54.21(3).

Duke Response to New Open Item 3.0.3.15.2-1
Duke understands that the staff's concern in this new open item is structural integrity of piping systems due to loss of material, in particular localized corrosion, under all design basis conditions.

The Service Water Piping Corrosion Program, formalized as a part of Duke's response to NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment," utilizes ultrasonic technology to look for loss of material which includes both general and localized corrosion. The program includes inspection locations representative of every pipe size, in each analysis model pipe run, for each flow regime, and upstream/downstream of each major piece of equipment. The periodic ultrasonic testing (UT) at these locations will identify any potential areas of severe degradation, including general and localized corrosion, which could exceed the ability of the piping to maintain its structural integrity in a design basis event. Inspection results are used to determine and expand, as necessary, the number of inspection locations in a given characteristic set.

As Duke has previously described, the primary issue addressed by the program is gross wall loss. Gross wall loss is deterioration of material condition sufficiently extensive to lead to structural instability and loss of component intended function. The secondary issue addressed by the program is the gathering of other symptomatic evidence that will serve as anomalous indications of material degradation. An example of such evidence is pinhole leaks caused by pitting and localized corrosion. As made clear by the Code design rules, pitting absent general corrosion is not a structural concern under normal operation or design basis conditions unless there exists a large number of pits in one area. A large number of pits in one area is essentially gross wall loss.

When an occurrence of localized corrosion is identified either through a low UT reading or a pinhole leak, an evaluation is performed to justify its structural integrity under all design basis conditions (in accordance with the appropriate design code and under guidance of NRC Generic Letter 90-05, "Guidance for Performing Temporary Non-code Repair of ASME Code Class 1, 2 and 3 Piping.") Additionally, UT will detect if there are numerous occurrences of localized corrosion in a given sample area because it does "look" like gross wall loss. As described in the Application, occurrences of localized corrosion are trended to assure an awareness of the progression of the material condition.

The staff concern that localized corrosion can lead to a structural integrity concern before it is revealed as pinhole leaks is valid. The Service Water Corrosion Program has been designed to address this concern by performing appropriate inspections, evaluations and trending and by taking appropriate corrective actions. The Service Water Corrosion Program is subject to ongoing regulatory oversight including the Service Water System Operational Performance Inspection (SWSOPI) and the Safety Systems Engineering Inspection (SSEI) McGuire and Catawba both have been inspected in recent years. This aging management program is consistent with the GALL Report and with the similar program at Oconee which the staff has found adequate for license renewal (Reference NUREG-1723, Section 3.2.13). As such the Service Water Corrosion Program can adequately manage loss of material from both general and localized corrosion for the license renewal systems that credit this program so that the component intended functions will be maintained consistent with the current licensing basis for the period of extended operation.

The **Monitoring & Trending** attribute of the summary description of the *Service Water Piping Corrosion Program* described in each station's UFSAR Supplement will be revised to read as follows:

Monitoring & Trending – The *Service Water Piping Corrosion Program* manages all of the system components within license renewal that are susceptible to the various corrosion mechanisms and is not focused on individual components within each specific system. The intent of the *Service Water Piping Corrosion Program* is to inspect a number of locations with conditions that are characteristic of the conditions found throughout the raw water systems above. The results of these inspection locations would then be applied to similar locations throughout all the raw water systems within the scope of license renewal. This characteristic-based approach recognizes the commonality among the component materials of construction and the environment to which they are exposed. Inspection results are used to determine and expand, as necessary, the number of inspection locations in a given characteristic set.

Monitoring under the *Service Water Piping Corrosion Program* focuses on carbon steel pipe. For components constructed of cast and ductile iron, galvanized steel and copper alloys, experience has shown that loss of material for these components will occur at a rate somewhat less than the carbon steel pipe. Therefore, the results of the carbon steel pipe inspections will provide a leading indicator of the condition of these materials.

For the carbon and galvanized steel, cast and ductile iron, and copper alloy component materials that can experience loss of material from both uniform and localized mechanisms, it is the gross material loss due to uniform mechanisms that is of primary concern under the *Service Water Piping Corrosion Program.* Gross wall loss can lead to structural instability concerns and could directly impact component intended function. Monitoring for degradation, including general and localized corrosion, is accomplished using ultrasonic test techniques. Monitoring for general and localized corrosion is supplemented by visual inspections of the inside of the piping if access to the interior surfaces is allowed such as during plant modifications. Monitoring of localized corrosion is additionally supplemented by exterior piping inspections that reveal pinhole leaks caused by localized corrosion. Additional detail concerning exterior piping inspections is provided below.

When pipe wall thickness is determined by volumetric wall thickness measurements using ultrasonic testing, several measurements are taken around the circumference of the piping. These measurements are then assessed in relation to the specific acceptance criteria for that location. Because the phenomena is slow-acting, inspection frequency varies for each location. The frequency of re-inspection depends on previous inspection results, calculated rate of material loss, piping analysis review, pertinent industry events, and plant operating experience. Refer to **Acceptance Criteria** for additional details. Component results are catalogued, and future inspection or component replacement schedules are determined as a part of the program.

Supplemental visual inspection detect localized corrosion due to pitting and microbiologically-influenced corrosion (MIC) that reveals itself through pinhole leaks in the piping components. The geometry of the pinholes means that they are not a structural integrity concern. Further, these pinhole leaks cannot individually lead to loss of the component intended function, since sufficient flow at prescribed pressures can still be provided by the system. These localized concerns will lead to structural integrity concerns only when a significant number of pinholes are present. When indications of a pinhole are found, volumetric wall thickness measurements are taken in the area. A trend of indications of through-wall leaks due to pitting corrosion or MIC provides evidence when localized corrosion may become a structural integrity concern and will trigger corrective actions by the *Service Water Piping Corrosion Program.* Methods in place to identify incidents of through-wall leaks are system walkdowns, operator rounds, system testing, and maintenance activities.

While the emphasis of the *Service Water Piping Corrosion Program* remains on potential areas of severe degradation, including general and localized corrosion, the management of loss of material due to localized corrosion of component materials exposed to raw water is supplemented by the monitoring and trending of relevant plant operating experience of non-structural, through-wall leaks identified during various plant activities.

New Open Item 3.3.6.2.1-1 In its response to RAI 2.3.3.6-6, the applicant provided the AMR results for condenser circulating water system expansion joints at Catawba. The material for these expansion joints was specified as synthetic rubber coated with chlorobutyl rubber; the environment was specified as the yard. The applicant did not identify any aging effects; nor did the applicant specify any AMP for these components. However, the staff concluded that exposure of these expansion joints to ultraviolet (UV) rays could cause degradation over time, Because the applicant's description of the yard environment in the LRA did not address sun exposure, the staff was unable to verify that there are no applicable aging effects for these components. The applicant needs to submit a more detailed description of the yard environment for the condenser circulating water system expansion joints to address UV exposure.

This open item was discussed with the staff during a meeting on September 18, 2002. By letter dated October 19, 2002, the staff requested information that is in addition to that contained in the above version of New Open Item 3.3.6.2.1-1.

Duke Response to New Open Item 3.3.6.2.1-1
In accordance with the requested response date provided in the October 19, 2002 staff letter, the Duke response to New Open Item 3.3.6.2.1-1 will be provided by November 6, 2002.

New Open Item 3.3.17.2.1-1 In its response to RAI 2.3.3.17-2, the applicant provided the AMR results for a carbon steel emergency diesel generator starting air distributor filter in a sheltered environment. The applicant indicated that no aging effects were identified for this component. However, the staff noted that this conclusion was not consistent with the applicant's treatment of other carbon steel components in a sheltered (moist air) environment that are listed in Table 3.3-23, "Aging Management Review Results - Diesel Generator Starting Air System (McGuire Nuclear Station)." The applicant needs to explain why the carbon steel emergency diesel generator starting air distributor filter in a sheltered environment is not subject to loss of material or identify this aging effect and an AMP to manage or monitor the associated loss of material.

Duke Response to New Open Item 3.3.17.2.1-1
The response to RAI 2.3.3.17-2 provided by Duke is in error. The carbon steel emergency diesel generator starting air distributor filter is subject to loss of material in a sheltered environment. The table entry provided in the response to RAI 2.3.3.17-2 should be replaced with the following:

Component Type	Component Function	Material	Internal Environment / External Environment	Aging Effect	Aging Management Programs and Activities
Starting Air Distributor Filter	Pressure Boundary	CS	Air (Dry)	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Open Item 3.3.35.2-1 The staff requested additional information pertaining to Table 3.3-44, "Aging Management Review Results - Standby Shutdown Diesel Generator." This table indicates that the cooling water and jacket water engine radiator heat exchanger has a heat transfer function that is managed by the Chemistry Control Program. Heat transfer monitoring is not identified as a capability of the Chemistry Control Program, as defined in Appendix B, Section B.3.6. The applicant was requested to explain how the Chemistry Control Program monitors the heat transfer function. In its response, the applicant stated that for the heat exchangers in the standby shutdown diesel generator cooling water and jacket water heating sub-system, fouling would not occur because there is constant flow through the heat exchangers and because the treated water in the system is filtered to remove particles. Therefore, no aging management program is required. The staff does not agree with the applicant's conclusion that fouling will not occur in the heat exchanger because of the constant flow through the heat exchanger. The staff recognizes that sufficient flow through the heat exchanger may prevent areas of stagnation in which fouling may occur. However, the applicant has not substantiated its conclusion with any operating experience, such as maintenance and surveillance results, that reflect the success of this activity in preventing fouling. With respect to the filtering of the treated water to remove particles, the staff recognizes that particulates are removed through a filtering process. However, the applicant did not list or credit a periodic surveillance of the filter to ensure that the entrained particles do not create a high differential pressure and adversely affect flow through the heat exchanger.

Duke Response to Open Item 3.3.35.2-1
Duke will identify fouling due to silting as an aging effect requiring management for the heat exchanger in the Standby Shutdown Diesel Cooling Water and Jacket Water Heating Subsystem that is managed by the *Chemistry Control Program.* Fouling due to silting is the result of corrosion products being generated throughout the system and deposited in the heat exchanger. The Standby Shutdown Diesel Cooling Water and Jacket Water Heating Subsystems are closed cooling water systems treated with corrosion inhibitors. The corrosion inhibitors preclude the formation of corrosion products. The corrosion inhibitor concentration in the system is monitored by the Chemistry Control Program. The Chemistry Control Program manages fouling due to silting during the period of extended operation by monitoring and maintaining the corrosion inhibitor concentration to preclude the formation of corrosion products.

The second entry in Table 3.3-44, Aging Management Review Results – Standby Shutdown Diesel, on page 3.3-247 of the Application should be replaced with the following:

Component Type	Component Function	Material	Internal Environment / External Environment	Aging Effects	Aging Management Programs and Activities
Heat Exchanger Engine Radiator (tubes)	Pressure Boundary, Heat Transfer	Copper	Treated Water	Loss of Material Fouling	Chemistry Control Program
			Ventilation	None Identified	None Required

Based on the evaluations provided in Appendix B for the aging management programs and activities listed above, the aging effects will be adequately managed such that the intended functions of the components listed in the above table will be maintained consistent with the current licensing basis for the period of extended operation.

New Open Item 3.4.1.2.2-1 The applicant proposes to mitigate general corrosion and loss of material of the auxiliary feedwater system carbon steel piping components by chemistry control. However, the staff believes that the effectiveness of the Chemistry Control program should be verified by implementing a one-time inspection of the internal surfaces of these components.

Duke Response to New Open Item 3.4.1.2.2-1
Duke disagrees with the staff that the effectiveness of *Chemistry Control Program* needs to be verified by a one-time visual inspection.

Section B.3.6 of Appendix B of the LRA provides a description of the *Chemistry Control Program.* The **Operating Experience** attribute on page B.3.6-4 provides the Duke specific experience to demonstrate the effectiveness of the *Chemistry Control Program* for managing aging effects. A search of the Problem Investigation Process database was performed to demonstrate the effectiveness of the *Chemistry Control Program.* Reports are entered into the database for component failures, relevant industry operating experience, and problems discovered during routine maintenance and testing. This review of operating experience did not reveal any instances of a loss of the component intended functions of the Auxiliary Feedwater System components that could be attributed to the inadequacy of the *Chemistry Control Program.*

Additionally, routine maintenance of other secondary side components such as the steam generators and main turbine provide additional operating experience because, although the Auxiliary Feedwater System is normally in standby, it does operate during startup and shutdown and is of the same chemistry as the Feedwater System and other secondary side systems. This good operating experience demonstrates the effectiveness of the *Chemistry Control Program* and does not warrant a one-time inspection.

During a meeting on September 18, 2002, the staff indicated that a commitment to provide written evidence of visual inspections of the Auxiliary Feedwater System and the Main Feedwater System that demonstrates that there are no aging effects occurring would be acceptable. Duke disagrees with the staff request and continues to believe that the existing *Chemistry Control Program* is well-managed and subject to periodic regulatory oversight. Any additional objective evidence of its effectiveness is unnecessary to make a reasonable assurance finding.

Nevertheless and as a practical matter in order to support the timely resolution of this open item and the completion of the license renewal review on schedule, Duke will not challenge this issue further.

The McGuire UFSAR Supplement, Section 18.3, will be revised to add the following commitment:

Visual inspections of the interior surfaces of Auxiliary Feedwater System and Main Feedwater System components and piping will be performed when available. The inspection results will be documented in writing and available for inspection following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1).

The Catawba UFSAR Supplement, Section 18.3, will be revised to add the following commitment:

Visual inspections of the interior surfaces of Auxiliary Feedwater System and Main Feedwater System components and piping will be performed when available. The inspection results will be documented in writing and available for inspection following issuance of renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba Unit 1).

Confirmatory Item 2.3.3.26.2-1 By letter dated January 28, 2002, the staff requested, in RAI 2.3.3.26-2, the applicant to indicate if piping and nitrogen cylinders associated with a safety-related backup nitrogen control system were within the scope of license renewal. In its response, dated April 15, 2002, the applicant confirmed that the Catawba main steam line PORVs are supplied with a nitrogen control system backup to the normal instrument air supply. This backup nitrogen control system consists of valves, tubing, and nitrogen bottles. The applicant stated that the nitrogen bottles are periodically replaced and, therefore, are not subject to an AMR. However, the applicant did not specify the details of the periodic replacement. In electronic correspondence dated July 16, 2002, the applicant stated that a Catawba technical specification surveillance procedure requires nitrogen cylinder replacement if the pressure in either nitrogen cylinder is less than or equal to 2420 psig. Pending the staff's receipt of this information in official correspondence, this item is confirmatory.

Duke Response to Confirmatory Item 2.3.3.26.2-1
In response to Confirmatory Item 2.3.3.26.2-1, Duke formally provides the following which had been sent originally by electronic communication on July 16, 2002:

> Catawba technical specification SR 3.7.4.1 applies to the main steam line PORV nitrogen bottles. This technical specification requires that once every 24 hours at least one of the nitrogen bottles on each SG PORV is verified to be pressurized ≥ 2100 psig. This surveillance requirement is performed by a Catawba procedure entitled "Procedure for Checking and Replacing Steam Generator PORV Nitrogen Cylinders and Setting Cylinder Regulators." There are two nitrogen cylinders per SG PORV. Initial pressure in the cylinder is ≥ 2500 psig. This procedure requires that if the pressure in either nitrogen cylinder is less than or equal to 2420 psig, then the nitrogen cylinder is replaced. Replacement cylinders are obtained from a warehouse. The used cylinders are returned to the warehouse. The cylinders are not permanently installed in the plant.

Attachment 3

Response to
McGuire Units 1 & 2 and Catawba Units 1 & 2
Safety Evaluation Report with Open Items

Information to Supplement Responses
Previously Provided by
Letter Dated October 2, 2002

Information in Response to Electronic
Communication of October 23, 2002

Open Item 2.5-1

During a meeting on September 17, 2002, Duke discussed with the staff its informal response to Open Item 2.5-1. As a result of the discussions on this open item, Duke agreed to provide a simplified drawing of the SBO recovery path for use by the staff during its then forthcoming meeting with the ACRS. Duke provided the requested information by electronic communication on September 24, 2002. By letter dated October 2, 2002, Duke provided its formal response to Open Item 2.5-1. Subsequently, the staff requested that the information provided via electronic communication be provided in docketed correspondence. Accordingly, the following is the information previously provided electronically:

ADDITIONAL INFORMATION REGARDING STATION BLACKOUT (SBO) SCOPING AND COMPONENT AGING MANAGEMENT REVIEWS

SBO Offsite Power Recovery Power Path – McGuire & Catawba

The power path for recovery of offsite power to each unit from the switchyard following an SBO starts with the generator busline power circuit breakers (PCBs) in the respective switchyards and includes the power connections through the generator buslines (transmission lines) to the main step-up transformers, through isolated-phase bus to the auxiliary transformers, through nonsegregated-phase bus to 6900 volt switchgear buses, and through cables to the 4160 volt safety-related buses.

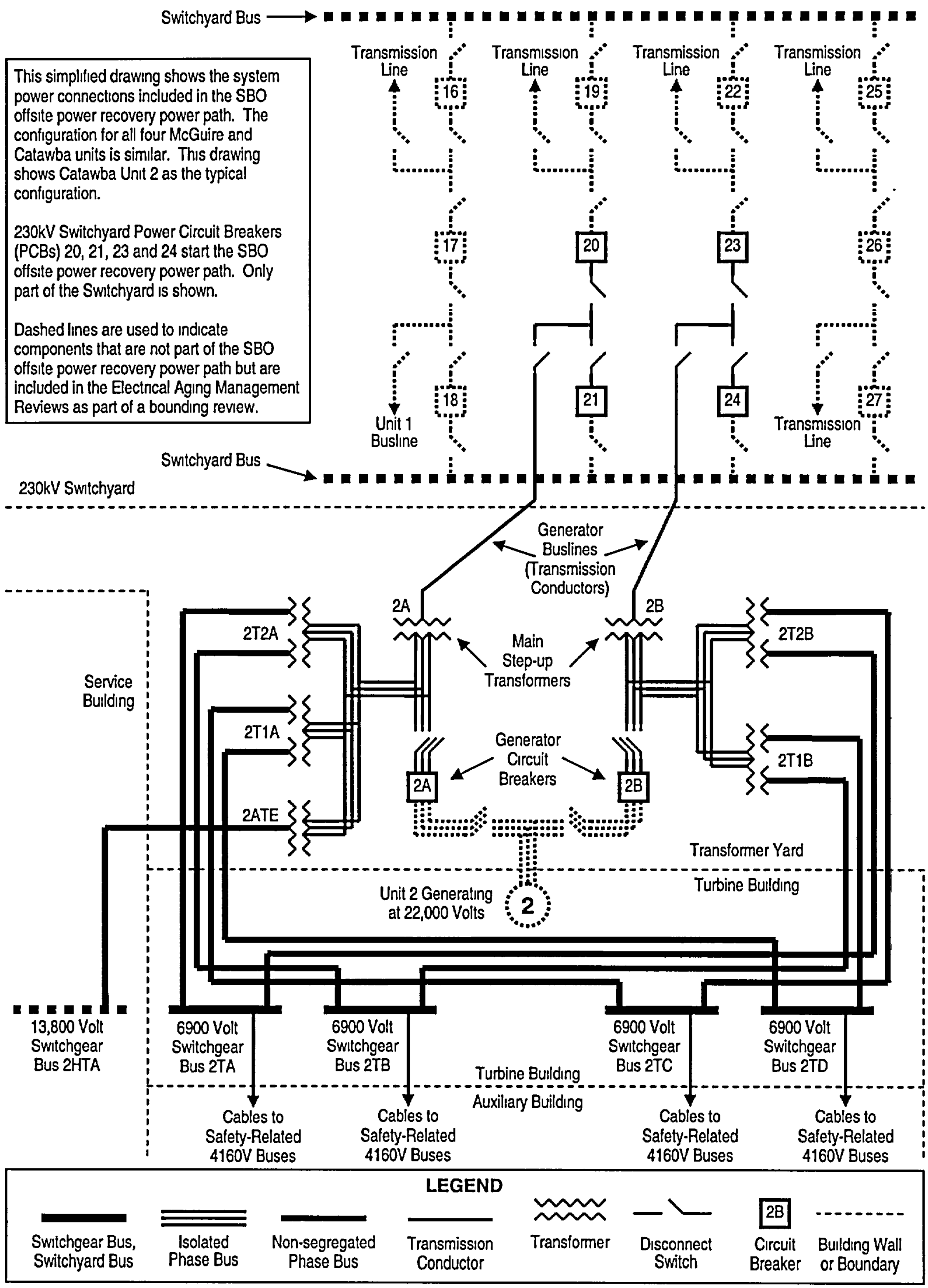
The simplified drawing on the next page shows these power connections. The configurations for all four McGuire and Catawba units are similar. The drawing shows Catawba Unit 2 as the typical configuration. A bounding scope of electrical components are included in the aging management reviews; e.g., even though only part of the switchyards are within scope for recovery of offsite power all passive electrical components in the switchyards are included in the aging management reviews. In the drawing dashed lines are used to indicate components that are not part of the SBO offsite power recovery power path but are included in the electrical aging management reviews as part of a bounding review.

General Design Criteria (GDC) 17 Offsite Power Connections

In conformance with GDC-17, reliability of offsite power to the station is assured by two separate and independent transmission lines per unit connecting the switchyard to the station. Both Catawba units connect to a 230kV switchyard. McGuire Unit 1 connects to a 230kV switchyard and McGuire Unit 2 connects to a 525kV switchyard. These two lines per unit supply power to two halfsized main step-up transformers which reduce the voltage to 22kV at Catawba and 24kV at McGuire. The use of two generator circuit breakers per unit allows immediate access to each of the preferred power sources. These sources maintain their independence within the auxiliary power system through separate voltage transformations from 22kV (or 24kV) to 6900V and then to 4160V. At the 4160V level these sources connect to and supply the Essential Auxiliary Power System. At McGuire or Catatawba there are no separate start-up transformers or other power connections to the switchyards.

Insulated Cables and Connections & SBO Recovery Scoping

In the June 2001 Application electrical scoping was performed and determined Switchyard Systems and the Unit Main Power Systems were not within scope. Following SBO offsite power recovery scoping these systems were scoped in. Due to the bounding approach taken for insulated cables and connections (i.e., no insulated cables and connections were scoped out), even though these systems were initially scoped out, the insulated cables and connections within these scoped-out systems were included in the June 2001 aging management review. No addition address of insulated cables and connections was needed due to the SBO offsite power recovery scoping since all insulated cables and connections were included in the initial review. There are no cables in the switchyards or installed elsewhere at McGuire and Catawba used in applications greater that 13.8kV.


This simplified drawing shows the system power connections included in the SBO offsite power recovery power path. The configuration for all four McGuire and Catawba units is similar. This drawing shows Catawba Unit 2 as the typical configuration.
230kV Switchyard Power Circuit Breakers (PCBs) 20, 21, 23 and 24 start the SBO offsite power recovery power path. Only part of the Switchyard is shown.
Dashed lines are used to indicate components that are not part of the SBO offsite power recovery power path but are included in the Electrical Aging Management Reviews as part of a bounding review.
Switchyard Bus
Transmission Line
Transmission Line
Transmission Line
Transmission Line
16
19
22
25
17
20
23
26
18
21
24
27
Unit 1 Busline
Transmission Line
Switchyard Bus
230kV Switchyard
Generator Buslines (Transmission Conductors)
2A
2B
2T2A
2T2B
Main Step-up Transformers
Service Building
2T1A
2T1B
Generator Circuit Breakers
2A
2B
2ATE
Transformer Yard
Turbine Building
Unit 2 Generating at 22,000 Volts
2
13,800 Volt Switchgear Bus 2HTA
6900 Volt Switchgear Bus 2TA
6900 Volt Switchgear Bus 2TB
6900 Volt Switchgear Bus 2TC
6900 Volt Switchgear Bus 2TD
Turbine Building
Auxiliary Building
Cables to Safety-Related 4160V Buses
Cables to Safety-Related 4160V Buses
Cables to Safety-Related 4160V Buses
Cables to Safety-Related 4160V Buses
LEGEND
Switchgear Bus, Switchyard Bus
Isolated Phase Bus
Non-segregated Phase Bus
Transmission Conductor
Transformer
Disconnect Switch
2B
Circuit Breaker
Building Wall or Boundary

Open Items 3.5-1 and 3.5-3

During a meeting on September 18, 2002, Duke discussed with the staff its informal response to Open Items 3.5-1 and 3.5-2. As a result of the discussions on these open items, Duke agreed to reconsider its informal response and consider committing to visually inspect concrete structural components. In its letter dated October 2, 2002, Duke agreed to credit its *Inspection Program for Civil Engineering Structures and Components* to manage the effects of aging for concrete structural components for the period of extended operation.

Subsequently, the staff in an electronic communication dated October 10, 2002 provided the following request:

> Please submit revised AMR results tables for all of Section 3.5, which should also include and clearly reference the concrete structures/components in the SBO recovery path that were brought into scope and for which no aging effects were identified. The revised tables must indicate the aging effect(s) for each structure or component as well as the AMP(s) credited.

In response to this supplemental staff request, Duke has developed this response based on information previously submitted to the staff. The original Tables 3.5-1, 3.5-2, and 3.5-3 contained in the Application have been revised:

(1) To indicate those structural components credited for SBO recovery as previously provided in Duke letter dated June 26, 2002. These structural components are indicated by an asterisk in the tables that follow. For completeness, both steel structural components and concrete structural components required for SBO recovery have be included.

(2) To indicate those concrete structural components which previously had no aging effects identified to include the aging effects which must be managed as well as the credited aging management programs. These revised concrete structural components are indicated in **BOLD** font to facilitate staff review.

Revised Tables 3.5-1, 3.5-2 and 3.5-3 are provided below.

Revised Table 3.5-1 Aging Management Review Results – Reactor Building

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material**	**Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Concrete Shield Building					
Dome	**2, 3, 6, 7**	**Concrete**	**Reactor Building**	**(Note 4)**	**Technical Specification SR 3.6.16.3 Visual Inspection**
			External	Change in Material Properties due to leaching	Technical Specification SR 3.6.16.3 Visual Inspection
Foundation Dowels (McGuire only)	2, 7	Steel	Concrete	None Identified	None Required
Foundation Mat	2, 7, 11	Concrete	Below Grade	None Identified	None Required
Shell Wall	**2, 3, 4, 6, 7, 11**	**Concrete**	**Reactor Building**	**(Note 4)**	**Technical Specification SR 3.6.16.3 Visual Inspection**
			Below Grade	None Identified	None Required
			External	Change in Material Properties due to leaching	Technical Specification SR 3.6.16.3 Visual Inspection

Revised Table 3.5-1 Aging Management Review Results – Reactor Building (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material**	**Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Steel Containment					
Bellows (Penetration)	1	Stainless	Reactor Building	Cracking	Containment Leak Rate Testing Program
Electrical Penetrations	1	Steel	Reactor Building	Loss of Material	Containment Inservice Inspection Plan – IWE Containment Leak Rate Testing Program
Equipment Hatch	1	Steel	Reactor Building	Loss of Material	Containment Inservice Inspection Plan – IWE Containment Leak Rate Testing Program
Fuel Transfer Tube Penetration	1	Steel	Reactor Building	Loss of Material	Containment Inservice Inspection Plan – IWE Containment Leak Rate Testing Program
Mechanical Penetrations	1	Steel	Reactor Building	Loss of Material	Containment Inservice Inspection Plan – IWE Containment Leak Rate Testing Program
Personnel Air Locks	1	Steel	Reactor Building	Loss of Material	Containment Inservice Inspection Plan – IWE Containment Leak Rate Testing Program
Steel Containment Vessel	1	Steel	Reactor Building	Loss of Material	Containment Inservice Inspection Plan – IWE Containment Leak Rate Testing Program

Revised Table 3.5-1 Aging Management Review Results – Reactor Building (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material**	**Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Ice Condenser Components					
Ice Baskets	2, 7	Galvanized Steel	Ice Condenser	Loss of Material	Ice Condenser Inspections
Lattice Frames & Support Columns	2, 7	Steel	Ice Condenser	Loss of Material	Ice Condenser Inspections
Lower Inlet Doors, Intermediate Deck Doors, Top Deck Doors	2, 3, 7	Steel	Ice Condenser Reactor Building	Loss of Material	Ice Condenser Inspections
Lower Support Structure	2, 7	Steel	Ice Condenser	Loss of Material	Ice Condenser Inspections
Wear Slab	**Note 3**				

Revised Table 3.5-1 Aging Management Review Results – Reactor Building (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material**	**Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Reactor Building Interior Structural Components					
Anchorage	2, 7, 11	Steel	Concrete	None Identified	None Required
Anchorage (exposed surface)	2, 7, 11	Steel	Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Checkered Plate	3	Steel	Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Embedments	2, 7, 11	Steel	Concrete	None Identified	None Required
Embedments (exposed surface)	2, 7, 11	Steel	Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Equipment Pads	**2, 7, 11**	**Concrete**	**Reactor Building**	**(Note 4)**	**Inspection Program for Civil Engineering Structures and Components**
Expansion Anchors	2, 7, 11	Steel	Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Revised Table 3.5-1 Aging Management Review Results – Reactor Building (continued)

1	2	3	4	5	6
Component Type (Note 1)	**Component Function (Note 2)**	**Material**	**Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Flood Curbs	**2, 8**	**Concrete**	**Reactor Building**	**(Note 4)**	**Inspection Program for Civil Engineering Structures and Components**
Flood Curbs (Steel)	2, 8	Steel	Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Revised Table 3.5-1 Aging Management Review Results – Reactor Building (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material**	**Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Reactor Building Interior Structural Components (continued)					
Flood, Pressure, & Specialty Doors	1, 3, 8	Steel	Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Fuel Transfer Canal Liner Plate	1	Stainless	Reactor Building	None Identified	None Required
Hatches	**3, 6, 11**	**Concrete**	**Reactor Building**	**(Note 4)**	**Inspection Program for Civil Engineering Structures and Components**
Missile Shields	**3, 6**	**Concrete**	**Reactor Building**	**(Note 4)**	**Inspection Program for Civil Engineering Structures and Components**
			External (equipment hatch missile shield)	**(Note 4)**	**Inspection Program for Civil Engineering Structures and Components**
Pressure Seals & Gaskets	1	EPDM (Note 2)	Reactor Building	Cracking Change in Material Properties	Divider Barrier Seal Inspection and Testing Program
Reinforced Concrete Beams, Columns, Floor Slabs, Walls	**1, 2, 3, 4, 6, 7, 8, 10, 11**	**Concrete**	**Reactor Building**	**(Note 4)**	**Inspection Program for Civil Engineering Structures and Components**

Revised Table 3.5-1 Aging Management Review Results – Reactor Building (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material**	**Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Reactor Building Interior Structural Components (continued)					
Structural Steel Beams, Columns, Plates & Trusses	2, 7, 11	Steel	Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Sump Liner	1	Stainless	Reactor Building	None Identified	None Required
Sump Screens (recirculation intake screen)	2	Stainless	Reactor Building	None Identified	None Required
Sumps	**2**	**Concrete**	**Reactor Building**	**(Note 4)**	**Inspection Program for Civil Engineering Structures and Components**

Notes for Revised Table 3.5-1 Aging Management Review Results – Reactor Building:

(1)	**Component Function**
1	Provides pressure boundary and/or fission product barrier.
2	Provides structural and/or functional support to safety-related equipment.
3	Provides shelter/protection to safety-related equipment.
4	Provides rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant.
5	Provides Ultimate Heat Sink following a LOCA or loss of Lake Norman or Lake Wylie.
6	Serves as missile (internal or external) barrier.
7	Provides structural and/or functional support to non-safety related equipment where failure of this component could directly prevent satisfactory accomplishment of any of the required safety-related functions.
8	Provides a protective barrier for internal/external flood event.
9	Provides path for release of filtered and unfiltered gaseous discharge.
10	Provides heat sink during SBO or design basis accidents.
11	Provides structural support and/or shelter to components relied on during certain postulated fire, anticipated transients without scram, and/or station blackout events.

(2) EPDM is the acronym for ethylene propylene dienyl monomer.

(3) Wear Slab entry in this revised Table 3.5-1 has been retained to maintain consistency with the original Table 3.5-1 provided in the Application. As discussed during a meeting with the staff on September 18, 2002, the wear slab no longer has an intended function. Response to Open Item 3.5-3 provided by Duke letter dated October 2, 2002 formally dockets this conclusion.

(4) Duke did not identify any aging effects that would result in loss of component intended function. The staff in its SER dated August 14, 2002 identified loss of material, cracking, and changes in material properties to be both plausible and applicable aging effects for all concrete components. Notwithstanding the disagreement on the aging effects that require management for the period of extended operation, Duke committed, in its response to Open Items 3.5-1 and 3.5-3 provided in a letter dated October 2, 2002, to perform periodic inspections of these concrete components using the *Inspection Program for Civil Engineering Structures and Components.*

Revised Table 3.5-2 Aging Management Review Results – Other Structures

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type (Note 1)	**Component Function (Note 2)**	**Material**	**Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Concrete Structural Components					
Equipment Pads *	**2, 7, 11**	**Concrete**	**Sheltered**	**(Note 4)**	**Inspection Program for Civil Engineering Structures and Components**
Fire Walls	4	Concrete	Sheltered	Cracking	Fire Protection Program
Flood Curbs	**8**	**Concrete**	**Sheltered**	**(Note 4)**	**Inspection Program for Civil Engineering Structures and Components**
Foundation Cassions (MNS TB only)	11	Concrete	Below Grade	None Identified	None Required
Foundations *	2, 7, 11	Concrete	Below Grade	None Identified	None Required
Hatches	**3, 4, 6, 11**	**Concrete**	**Sheltered**	**(Note 4)**	**Inspection Program for Civil Engineering Structures and Components**
Manholes & Covers (CNS NSW only)	**3**	**Concrete**	Below Grade	None Identified	None Required
			External	**(Note 4)**	**Inspection Program for Civil Engineering Structures and Components**

Revised Table 3.5-2 Aging Management Review Results – Other Structures (continued)

1	2	3	4	5	6
Component Type (Note 1)	**Component Function (Note 2)**	**Material**	**Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Missile Shields (AB and NSW only)	**3, 6**	**Concrete**	**Sheltered (AB only) External (AB and NSW only)**	**(Note 4)**	**Inspection Program for Civil Engineering Structures and Components**
			Raw Water (NSW only)	Loss of Material Cracking	Underwater Inspection of Nuclear Service Water Structures Inspection Program for Civil Engineering Structures and Components
Missile Shield (RWST Missile Shield Wall)	6	Concrete	External	Change in material properties due to leaching	Inspection Program for Civil Engineering Structures and Components

Revised Table 3.5-2 Aging Management Review Results – Other Structures (continued)

1	2	3	4	5	6
Component Type (Note 1)	**Component Function (Note 2)**	**Material**	**Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Concrete Structural Components (continued)					
Reinforced Concrete Beams, Columns, Floor Slabs, Walls *	**1(AB only), 2, 3, 4, 6, 7, 8 (AB only), 10 (AB only), 11**	**Concrete**	**Sheltered**	**(Note 4)**	**Inspection Program for Civil Engineering Structures and Components**
			External	Change in material properties due to leaching	Inspection Program for Civil Engineering Structures and Components
			Below Grade	None Identified	None Required
			Raw Water (NSW and LPSW (CNS))	Loss of Material Cracking	Inspection Program for Civil Engineering Structures and Components Underwater Inspection of Nuclear Service Water Structures
Roof Slabs	2, 3, 6, 7, 11	Concrete	External	Change in material properties due to leaching	Inspection Program for Civil Engineering Structures and Components
Sumps (AB only)	**1, 2**	**Concrete**	**Sheltered**	**(Note 4)**	**Inspection Program for Civil Engineering Structures and Components**
Trenches (Yard only) *	3, 11	Concrete	Below Grade	None Identified	None Required

Revised Table 3.5-2 Aging Management Review Results – Other Structures (continued)

1	2	3	4	5	6
Component Type (Note 1)	**Component Function (Note 2)**	**Material**	**Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Steel Structural Components					
Anchorage *	2, 7, 11	Steel	Concrete	None Identified	None Required
Anchorage (exposed surface) *	2, 7, 11	Steel	Sheltered	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components
			External/Raw Water	Loss of Material	Inspection Program for Civil Engineering Structures and Components Underwater Inspection of Nuclear Service Water Structures
Checkered Plate *	3, 11	Steel	Sheltered External	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components
Embedments *	2, 7, 11	Steel	Concrete	None Identified	None Required
Embedments (exposed surface) *	2, 7, 11	Steel	Sheltered External (Yard only)	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components

Revised Table 3.5-2 Aging Management Review Results – Other Structures (continued)

1	2	3	4	5	6
Component Type (Note 1)	**Component Function (Note 2)**	**Material**	**Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Steel Structural Components (continued)					
Expansion Anchors *	2, 7, 11	Steel	Sheltered External (Yard only)	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components
Fire Doors (AB and CNS NSW only)	4	Steel	Sheltered External (Yard only)	Loss of Material	Fire Protection Program
Flood Curbs	8	Steel	Sheltered	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components
Flood, Pressure, & Specialty Doors (AB, TB, and CNS NSW only)	1, 3, 8	Steel	Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Foundation Dowels (MNS AB and CCW only)	2, 7, 11	Steel	Concrete	None Identified	None Required
			Below Grade	None Identified	None Required
Metal Siding (MNS Battery Rooms only)	1, 3	Steel	Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Revised Table 3.5-2 Aging Management Review Results – Other Structures (continued)

1	2	3	4	5	6
Component Type (Note 1)	**Component Function (Note 2)**	**Material**	**Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Steel Structural Components (continued)					
Roof (MNS Fire Pump enclosure roof cover)	11	Steel	External	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Spent Fuel Pool Liner Plate (AB only)	1, 3	Stainless	Borated Water	Loss of Material Cracking	Chemistry Control Program
Structural Steel Beams, Columns, Plates & Trusses *	2, 7, 11	Steel	Sheltered External (Yard only)	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components
Structural Steel and Plates	2, 7	Stainless	Borated Water (AB only)	Loss of Material Cracking	Chemistry Control Program
			External/Raw Water (NSW only)	Loss of Material	Underwater Inspection of Nuclear Service Water Structures Inspection Program for Civil Engineering Structures and Components

Revised Table 3.5-2 Aging Management Review Results – Other Structures (continued)

Component Type (Note 1)	Component Function (Note 2)	Material	Environment	Aging Effects	Aging Management Programs and Activities
Steel Structural Components (continued)					
Sump Screens (AB only)	2	Steel	Sheltered	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components
Trash Rack and Screens (NSW only)	2	Stainless or Steel (CNS only)	Raw Water	Loss of Material	Underwater Inspection of Nuclear Service Water Structures
Unit Vent Stack	9	Steel	Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
			External	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Yard Drainage System (CNS only)	7	Steel	External	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Revised Table 3.5-2 Aging Management Review Results – Other Structures (continued)

1	2	3	4	5	6
Component Type (Note 1)	**Component Function (Note 2)**	**Material**	**Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Other Structural Components					
Boraflex Panels (MNS AB only)	2, 7	Boraflex	Borated Water	Degradation due to Gamma irradiation	Boraflex Monitoring Program
Earthen Embankment	2, 5	Soil	External	Loss of Material Cracking	Standby Nuclear Service Water Pond Dam Inspection
Fire Barrier Penetration Seals (AB and CNS NSW only)	4	Silicone	Sheltered	Cracking Separation	Fire Protection Program
		Rubber	Sheltered	Cracking	Fire Protection Program
Flood Seals	8	Rubber Silicon	Sheltered	Cracking Change in Material Properties	Flood Barrier Inspection (MNS only) Inspection Program for Civil Engineering Structures and Components (CNS Only)
Masonry Block Walls (AB, SSF, TB only) *	2, 3, 4, 7, 11	Masonry	Sheltered	Cracking	Inspection Program for Civil Engineering Structures and Components
Metal Siding (Yard only) *	3	Aluminum	External	None Identified	None Required
Roofing *	3, 11	Composite	External	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Notes for Revised Table 3.5-2 Aging Management Review Results – Other Structures:

(1) Location Abbreviations

AB = Auxiliary Building
CCW = Condenser Cooling Water Intake Structure (McGuire Fire Pump Rooms only)
CNS = Catawba Nuclear Station
LPSW = Low Pressure Service Water Intake Structure (Catawba)
MNS = McGuire Nuclear Station
NSW = Nuclear Service Water Structures
RB = Reactor Building
SNSWP = Standby Nuclear Service Water Pond Dam
SSF = Standby Shutdown Facility
TB = Turbine Buildings
* = An asterisk denotes that the Component Type is part of the SBO recovery path as identified in Duke letter dated June 26, 2002 to the NRC staff

(2) Component Function

1 Provides pressure boundary and/or fission product barrier.
2 Provides structural and/or functional support to safety-related equipment.
3 Provides shelter/protection to safety-related equipment.
4 Provides rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant.
5 Provides Ultimate Heat Sink following a LOCA or loss of Lake Norman or Lake Wylie.
6 Serves as missile (internal or external) barrier.
7 Provides structural and/or functional support to non-safety related equipment where failure of this component could directly prevent satisfactory accomplishment of any of the required safety-related functions.
8 Provides a protective barrier for internal/external flood event.
9 Provides path for release of filtered and unfiltered gaseous discharge.
10 Provides heat sink during SBO or design basis accidents.
11 Provides structural support and/or shelter to components relied on during certain postulated fire, anticipated transients without scram, and/or station blackout events. (For McGuire CCW and SSF, Function 11 only applies; For Catawba LPSW and SSF, Function 11 only applies.)

(3) The Fluid Leak Management Program is applicable for structural components that are listed in this table that are only located in the Auxiliary Building.

(4) Duke did not identify any aging effects that would result in loss of component intended function. The staff in its SER dated August 14, 2002 identified loss of material, cracking, and changes in material properties to be both plausible and applicable aging effects for all concrete components. Notwithstanding the disagreement on the aging effects that require management for the period of extended operation, Duke committed, in its response to Open Items 3.5-1 and 3.5-3 provided in a letter dated October 2, 2002, to perform periodic inspections of these concrete components using the *Inspection Program for Civil Engineering Structures and Components.*

Revised Table 3.5-3 Aging Management Review Results – Component Supports

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type (Note 1)	**Component Function (Note 2)**	**Material**	**Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Battery Racks (AB, SSF only) *	2, 11	Steel	Sheltered	Loss of Material	Battery Rack Inspections
Cable Tray & Conduit *	2, 7, 11	Steel	Sheltered Reactor Building External (Yard only)	None Identified	None Required
Cable Tray & Conduit Supports *	2, 7, 11	Steel	Sheltered Reactor Building External (Yard only)	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components
Class 1 (NSSS) Supports	2	Steel	Reactor Building	Loss of Material	Fluid Leak Management Program (Note 3) Inservice Inspection Plan – Subsection IWF Inspection Program for Civil Engineering Structures and Components
Control Boards (AB, SSF only)	2, 3, 7, 11	Steel	Sheltered	None Identified	None Required
Control Room Ceiling (AB only)	7	Steel	Sheltered	None Identified	None Required

Revised Table 3.5-3 Aging Management Review Results – Component Supports (continued)

1	2	3	4	5	6
Component Type (Note 1)	**Component Function (Note 2)**	**Material**	**Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Crane Rails & Girders (AB, RB only)	7	Steel	Sheltered Reactor Building	Loss of Material	Crane Inspection Program
Electrical & Instrument Panels & Enclosures *	2, 3, 7, 11	Steel	Sheltered Reactor Building External (Yard only)	None Identified	None Required
Equipment Component Supports *	2, 7, 11	Steel	Sheltered Reactor Building External (Yard only)	Loss of Material	Inservice Inspection Plan – Subsection IWF Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components
			Raw Water (NSW only)	Loss of Material	Underwater Inspection of Nuclear Service Water Structures Inspection Program for Civil Engineering Structures and Components
		Galvanized Steel	Exterior	None Identified	None Required

Revised Table 3.5-3 Aging Management Review Results – Component Supports (continued)

1	2	3	4	5	6
Component Type (Note 1)	**Component Function (Note 2)**	**Material**	**Environment**	**Aging Effects**	**Aging Management Programs and Activities**
HVAC Duct Supports (RB, AB, SSF, and CNS NSW only) *	2, 7, 11	Steel	Sheltered Reactor Building	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components
Instrument Racks & Frames *	2, 7, 11	Steel	Sheltered Reactor Building	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components
Instrument Line Supports	2, 7, 11	Steel	Sheltered Reactor Building	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components
Lead Shielding Supports (RB and AB only)	7	Steel	Sheltered Reactor Building	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components
New Fuel Storage Racks (AB only)	2	Steel	Sheltered	None Identified	None Required

Revised Table 3.5-3 Aging Management Review Results – Component Supports (continued)

1	2	3	4	5	6
Component Type (Note 1)	**Component Function (Note 2)**	**Material**	**Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Pipe Supports	2, 7, 11	Steel	Sheltered Reactor Building External (Yard only)	Loss of Material	Inservice Inspection Plan – Subsection IWF Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components
Spent Fuel Storage Racks (AB only)	2	Stainless	Borated Water	Loss of Material Cracking	Chemistry Control Program
Stair, Platform, and Grating Supports *	2, 7, 11	Steel	Sheltered Reactor Building External (Yard only)	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components
Transmission Towers (MNS Only) *	11	Galvanized Steel	External	None Identified	None Required

Notes for Revised Table 3.5-3 Aging Management Review Results – Component Supports:

(1) Location Abbreviations

AB = Auxiliary Building
CCW = Condenser Cooling Water Intake Structure (McGuire Fire Pump Rooms only)
CNS = Catawba Nuclear Station
MNS = McGuire Nuclear Station
NSW = Nuclear Service Water Structures
RB = Reactor Building
SNSWP = Standby Nuclear Service Water Pond Dam
SSF = Standby Shutdown Facility
TB = Turbine Buildings
* = An asterisk denotes that the Component Type is part of the SBO recovery path as identified in Duke letter dated June 26, 2002 to the NRC staff

(2) Component Function

1 Provides pressure boundary and/or fission product barrier.
2 Provides structural and/or functional support to safety-related equipment.
3 Provides shelter/protection to safety-related equipment.
4 Provides rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant.
5 Provides Ultimate Heat Sink following a LOCA or loss of Lake Norman or Lake Wylie.
6 Serves as missile (internal or external) barrier.
7 Provides structural and/or functional support to non-safety related equipment where failure of this component could directly prevent satisfactory accomplishment of any of the required safety-related functions.
8 Provides a protective barrier for internal/external flood event.
9 Provides path for release of filtered and unfiltered gaseous discharge.
10 Provides heat sink during SBO or design basis accidents.
11 Provides structural support and/or shelter to components relied on during certain postulated fire, anticipated transients without scram, and/or station blackout events. (For McGuire CCW and SSF, Function 11 only applies; For Catawba LPSW and SSF, Function 11 only applies.)

(3) The Fluid Leak Management Program is applicable for component supports only in the Auxiliary Building and the Reactor Building

Region II Aging Management Review Inspection Item

NRC Inspection Report 50-369/02-06, 50-370/02-06, 50-413/02-06 and 50-414/02-06 (at page 2 of the report) dated September 9, 2002 identified that the Inservice Inspection Plan does not include two Reactor Coolant System components even though the Inservice Inspection Plan was credited to manage the aging of these components within the Application. The components are the pressurizer surge and spray nozzle thermal sleeves and the steam generator divider plates.

Immediately following this inspection, Duke re-reviewed these components against the license renewal scoping criteria contained in §54.4 and determined that neither of these components meets any of these criteria. Duke is revising the in-house licensing basis engineering documents to reflect the results of this review.

On October 21, 2002 the license renewal project manager called and requested Duke describe what actions it had taken in response to this discrepancy identified in the inspection report. Duke provided its response informally. Subsequently, the staff requested the following by electronic communication on October 23, 2002:

> After discussing this information with Sam, PT and Frank, I have been informed that this constitutes a change to your application for license renewal that needs to be formally communicated to the staff. Please send a letter explaining the intent to exclude these components from license renewal and the technical basis for the determination that they do not meet any of the scoping criteria specified in 10 CFR 54.4. Additionally, it occurs to me that the ice condenser wear slab (which was discussed in the September 18 meeting with the staff) should be addressed as well, since it similarly constitutes a change to the application.

In response to this request, Duke provides the following information: With respect to the pressurizer surge and spray nozzle thermal sleeves, Table 3.1-1 of the Application groups the thermal sleeves with the nozzles. The nozzles perform a Reactor Coolant System pressure boundary function (§54.4(a)(1)(i)); the thermal sleeves do not. Aging management programs for these nozzles include Inservice Inspection Plan, Chemistry Control Program, Alloy 600 Aging Management Review, and of course the nozzles are within the Thermal Fatigue Management Program discussed in Chapter 4 of the Application. Duke is revising its in-house license renewal engineering specifications to correct the discussion of the thermal sleeves to state that they are not in scope because they do not perform a license renewal function.

The steam generator (SG) divider plate is located in the lower head of each SG and separates the hot leg primary fluid from the cold leg primary fluid. Reactor coolant is located on both sides of the SG divider plate. Clearly it does not perform any function required by §54.4. The Application incorrectly called this a pressure boundary function. The Inspection Report correctly noted that the Inservice Inspection Plan does not include this component within the scope of inspections. Duke is revising its in-house license renewal engineering specifications to correct the discussion of the divider plate to state that it is not in scope because it does not perform a license renewal function as defined by §54.4.

Changes to the in-house license renewal engineering specifications are being made in accordance with the Duke QA program.

In response to the staff's last comment concerning the ice condenser wear slab and its removal from the scope of license renewal, Duke has previously addressed this component in its response to Open Item 3.5-3 provided by letter dated October 2, 2002 (see Attachment 3, page 8).